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Revision 3

Light Water Reactor Sustainability Program

Risk-Informed Safety Margins Characterization (RISMC) Pathway Technical Program Plan

September 2015

DOE Office of Nuclear Energy

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Risk-Informed Safety Margins Characterization (RISMC) Pathway Technical Program Plan

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EXECUTIVE SUMMARY

Safety is central to the design, licensing, operation, and economics of nuclear power plants (NPPs). As the current Light Water Reactor (LWR) NPPs age beyond 60 years, there are possibilities for increased frequency of systems, structures, and components (SSCs) degradations or failures that initiate safety-significant events, reduce existing accident mitigation capabilities, or create new failure modes. Plant designers commonly “over-design” portions of NPPs and provide robustness in the form of redundant and diverse engineered safety features to ensure that, even in the case of well-beyond design basis scenarios, public health and safety will be protected with a very high degree of assurance. This form of defense-in-depth is a reasoned response to uncertainties and is often referred to generically as “safety margin.” Historically, specific safety margin provisions have been formulated, primarily based on “engineering judgment.”

The ability to better characterize (i.e., describe and quantify) safety margin is important to improved decision making about LWR design and long-term operation. A systematic approach to characterizing safety margins and the subsequent risk informed margins management options represents a vital input to the licensee and regulatory analysis and decision making that will be involved. In addition, as research and development (R&D) in the LWRS Program and other collaborative efforts yield new data and improved scientific understanding of physical processes that govern the aging and degradation of plant SSCs (and concurrently support technological advances in nuclear reactor fuels and plant instrumentation and control systems), needs and opportunities to better optimize plant safety, economics, and performance will become known.

The purpose of the Risk-Informed Safety Margins Characterization (RISMC) Pathway R&D is to support plant decisions for risk-informed margins management with the aim to improve economics, reliability, and sustain safety of current NPPs over periods of extended plant operations. The goals of the RISMC Pathway are twofold: (1) develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by NPP decision makers as part of risk-informed margin management strategies; (2) create an advanced RISMC Toolkit that enables more accurate representation of NPP safety margins. Included in this Toolkit are the next generation reactor systems-analysis code (RELAP-7); a probabilistic-based scenario simulation code (RAVEN); and a component aging and damage evolution mechanism simulation application (Grizzly). The RISMC methodology can optimize plant safety and performance by incorporating plant impacts, physical aging, and degradation processes into the safety analysis.

The methods and tools provided by RISMC are essential to a comprehensive and integrated risk-informed margin management approach that supports effective preservation of margin for both active and passive SSCs. The deliverables provided by the Pathway include: (1) reports describing the technical basis for risk-informed margins management and (2) the RISMC Toolkit. These deliverables will serve to provide a comprehensive approach and software to support safety-, reliability-, and economic-decisions needed for near- and long-term NPP operation.

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ACRONYMS

ASAMPSA_E	Advanced Safety Assessment Methodologies: Extended PSA
BWR	boiling water reactor
CASL	Consortium on Advanced Simulation of LWRs
CNWG	Civil Nuclear Energy Research and Development Working Group
CRA	computational risk assessment
CSNI	Committee on the Safety of Nuclear Installations
CWO	core-wide oxidation
DG	diesel generator
DOE	Department of Energy
ECR	equivalent cladding reacted
EPRI	Electric Power Research Institute
FLEX	Diverse and Flexible Coping Strategies
GUI	graphical user interface
HPC	High Performance Computing
IA	Industry Applications
INL	Idaho National Laboratory
LOCA	loss-of-coolant accident
LWR	light water reactor
MOOSE	Multiphysics Object-Oriented Simulation Environment
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NEET	Nuclear Energy Enabling Technologies
NEUP	Nuclear Energy University Program
NPP	nuclear power plants
NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development

PCT	peak cladding temperature
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PWR	pressurized water reactor
R&D	research and development
RAVEN	Risk Analysis in a Virtual Environment
RELAP-7	Reactor Excursion and Leak Analysis Program-7
RIMM	Risk-Informed Margin Management
RISMC	Risk-Informed Safety Margins Characterization
SBO	station black out
SSC	systems, structures, and component
T-H	thermal-hydraulics
WGEV	Working Group on Natural External Hazards

Risk-Informed Safety Margins Characterization (RISMC) Pathway Technical Program Plan

1. BACKGROUND

Safety is central to the design, licensing, operation, and economics of nuclear power plants (NPPs). As the current light water reactor (LWR) NPPs age beyond 60 years, there are possibilities for increased frequency of systems, structures, and components (SSC) degradations or failures that initiate safety-significant events, reduce existing accident mitigation capabilities, or create new failure modes. Plant designers commonly "over-design" portions of NPPs and provide robustness in the form of redundant and diverse engineered safety features to ensure that, even in the case of well-beyond design basis scenarios, public health and safety will be protected with a very high degree of assurance. This form of defense-in-depth is a reasoned response to uncertainties and is often referred to generically as "safety margin." Historically, specific safety margin provisions have been formulated primarily based on "engineering judgment." Further, these historical safety margins have been set conservatively (for example in design and operational limits) in order to compensate for uncertainties.

The LWR Sustainability program is focused on ensuring the safety and performance of the nuclear fleet to enhance operation efficiencies of existing plants, support long term operation of these plants, and provide confidence for subsequent license renewals. Within this Program, the Risk-Informed Safety Margins Characterization (RISMC) Pathway is solving technical issues for several of the "sustainability" dimensions that exist, as illustrated in Figure 1-1.

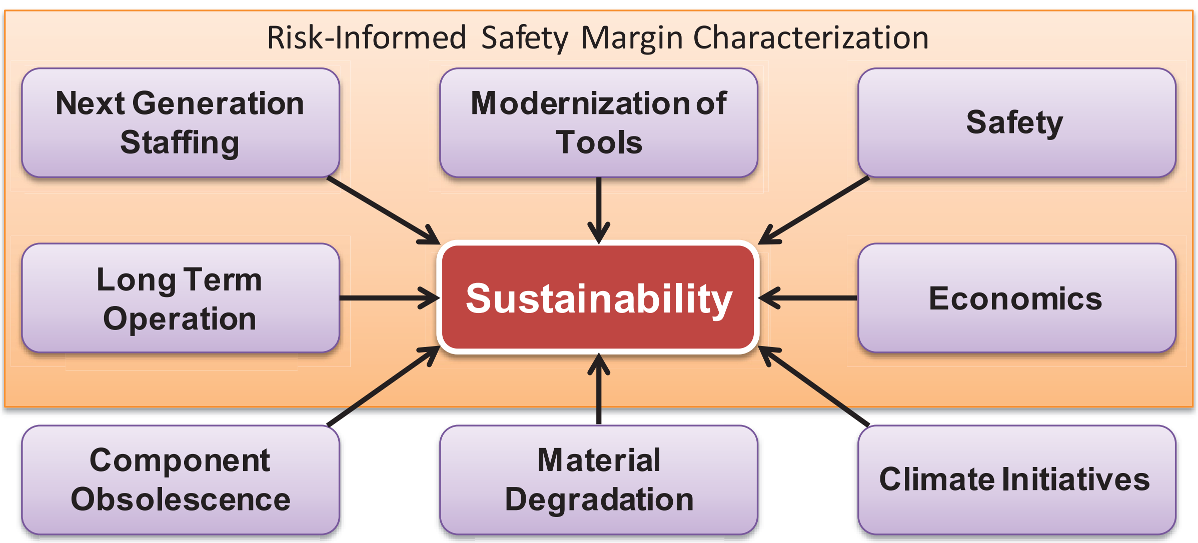

Figure 1-1. The sustainability aspects that exist for near and long term NPP operations.

Since safety is important to successful operation of the NPP fleet, there are strong motivations to better manage safety and its associated "margin." These motivations include having improved knowledge of both the qualitative and quantitative aspects of safety margins in order to provide for enhancements and improvement in NPPs, including support for applications such as:

- **Plant design changes.** During the NPP lifetime, plant changes are implemented following appropriate application of regulatory and licensing processes. For example, NPP extended power uprates may increase the plant production of power by a significant amount.

- **Operability issues**. During NPP operations, a variety of off-normal situations may arise such as licensing issues (e.g., nearing a limit for an allowable outage time) to failures of SSCs. Having an improved safety technical basis may provide an enhanced operational record (e.g., not having to shut down the plant) or a reduction in regulatory actions.

- **Addressing beyond design basis accidents**. As a result of the Fukushima event, the NRC established a task force to conduct a review of NRC processes and regulations to determine if the agency should make additional improvements to its regulatory system. This task force, known as the near-term task force gave its recommendations to the Commission in its report SECY-11-0137. [1] Currently, design basis requirements for NPP related to hazards such as flooding and seismic are primarily deterministic. However, the NRC's requests to the licensees may require insights both within and outside their design bases, thereby prompting the NRC to evaluate this information using improved methods such as safety margins in order to determine whether the design basis must be changed.

- **Plant life beyond sixty years**: The ability to better characterize (i.e., describe and quantify) safety margin provides a mechanism to improved decision making about LWR design, economics, operation, and long-term operation.

The RISMC methodology can optimize plant safety and performance by incorporating plant impacts, physical aging, and degradation processes into the safety analysis. A systematic approach to the characterization of safety margins and the subsequent margins management options represents a vital input to the licensee and regulatory analysis and decision making that will be involved. In addition, as R&D in the LWRS Program and other collaborative efforts yield new data and improved scientific understanding of physical processes that govern the aging and degradation of plant SSCs (and concurrently support technological advances in nuclear reactor fuels and plant instrumentation and control systems) needs and opportunities to better optimize plant safety and performance will become known. This interaction of improved understanding and potential impacts to plant margins is shown in Figure 1-2. To support decision making related to economics, reliability, and safety, the RISMC Pathway will provide methods and tools that enable mitigation options known as risk-informed margins management strategies.

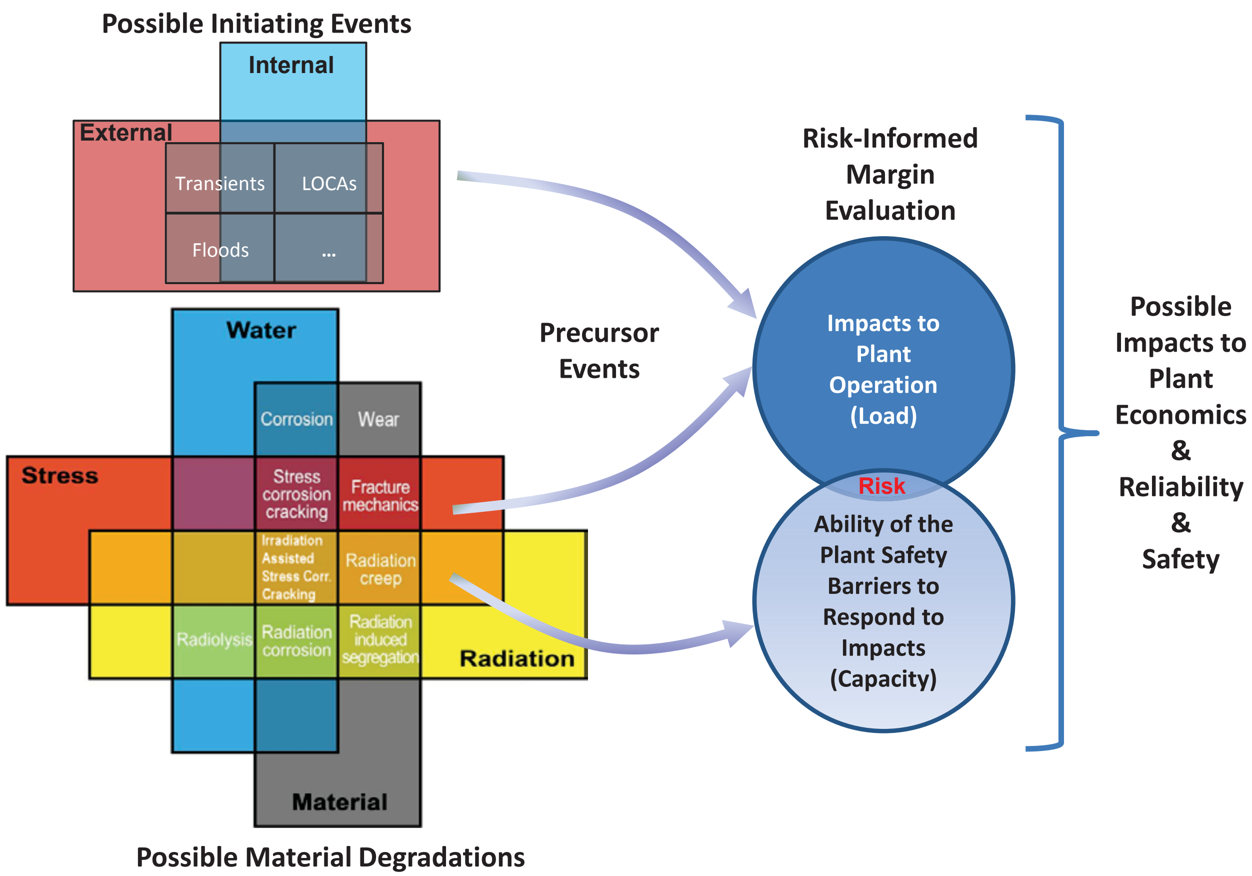

Figure 1-2. Representation of the interaction of degradation mechanisms that may impact plant operations and safety barriers if left unmitigated (adapted from INL 2012).

2. RESEARCH AND DEVELOPMENT

2.1 Purpose and Goals

The purpose of the RISMC Pathway R&D is to support plant decisions for Risk-Informed Margins Management with the aim to improve economics, reliability, and sustain safety of current NPPs over periods of extended plant operations.

The goals of the RISMC Pathway are twofold:

1. Develop and demonstrate a risk-assessment method that is coupled to safety margin quantification that can be used by NPP decision makers as part of risk-informed margin management strategies.

2. Create an advanced RISMC Toolkit that enables more accurate representation of NPP safety margins and their associated impacts on operations and economics.

One of the primary items inherent in the goals of the Pathway is the ability to propose and evaluate margin management strategies. If a situation exists that causes margins associated with one or more safety functions to become degraded, the methods and tools developed in this Pathway will serve to model and measure margins for active and passive SSCs for normal and off-normal conditions. These evaluations will then support development and evaluation of appropriate alternative strategies for consideration by decision makers to maintain and enhance the impacted margins as necessary. When alternatives are proposed that mitigate reductions in the safety margin, these changes are referred to as margin *recovery* strategies. Moving beyond current limitations in safety analysis, the Pathway will develop techniques to conduct margins analysis using simulation-based studies of safety margins.

Margin Management Strategies

Proposed alternatives (i.e., changes to SSCs or plant procedures) that work to control margin changes due to aging or plant modifications. Alternatives that off-set, or mitigate, reductions in the safety margin are known as margin recovery strategies.

While simulation methods in risk and reliability applications have been proposed for several decades, the availability of advanced mechanistic and probabilistic simulation tools have been limited. But, as noted by researchers such as Zio, [2] "…simulation appears to be the only feasible approach to quantitatively capture the realistic aspects of the multi-state system stochastic behavior." Consequently, the approach we are using for the RISMC Pathway is to use simulation tools to model plant behavior and determining safety margins. Specifically, we are developing the simulation components of the "RISMC Toolkit" which include:

- **RELAP-7:** The new generation nuclear reactor system safety analysis code is RELAP-7. The code is based upon the High Performance Computing (HPC) development and runtime framework – MOOSE (Multi-Physics Object-Oriented Simulation Environment). RELAP-7 will become the main reactor systems simulation tool for LWRS/RISMC. The design goal of RELAP-7 development is to leverage 30 years of advancements in software design, numerical integration methods, and physical models in order to seamlessly work in a risk analysis approach (e.g., to support analysis of many different possible scenarios) on a variety of computational resources (e.g., from laptops to computers with tens-of-thousands of processing cores).

- **RAVEN:** RAVEN (Risk Analysis in a Virtual Control Environment) is a multi-tasking application focused on simulation control, plant control logic, system analysis, uncertainty quantification, and scenario-generation for computational risk assessment (CRA) for postulated events. RAVEN has the capability to "drive" RELAP-7 (and other MOOSE-based applications) for which the following functional capabilities are provided:

    - Front-end driver for RELAP-7:
        - Input a plant description to RELAP-7 (component, control variable, and control parameters)
        - Runtime environment
        - Parallel distribution of RELAP-7 runs (adaptive sampling)
    - Control logic required to:
        - Simulate the reactor plant control system
        - Simulate the reactor operator (procedure guided) actions
        - Perform Monte Carlo sampling of stochastic events
        - Perform off-normal and accident-sequence based analysis
    - Control of Graphical User Interface (GUI) to:
        - GUI capability provided by Peacock (see below)
        - Concurrent monitoring of control parameters
        - Concurrent alteration of control parameters
    - Post-processing data mining capability based on:
        - Dimensionality reduction
        - Cardinality reduction
        - Uncertainty quantification and propagation

- **Peacock:** A graphical user interface for MOOSE that can be used to create, control, and interact with the various tools in the RISMC Toolkit.

- **Grizzly:** A MOOSE-based tool is being constructed for simulating component ageing and damage evolution events for LWRS specific applications. This new simulation tool, called Grizzly, will have implicit time simulation capabilities for component damage evolution concerning LWR pressure vessel, core internals, and concrete support and containment structures subjected to a neutron flux, corrosion, and high temperatures and pressures.

The RISMC Toolkit is being built using MOOSE, a computer simulation framework that simplifies the process for modeling complicated physics as represented by mechanistic models [3]. The MOOSE framework was developed by INL by using existing computer code and numerical libraries from proven scalable numerical tools developed at universities and DOE. The result is a framework with a number of high-level features that includes built-in parallelization and advanced geometry meshing capabilities. The constituent pieces of the overall RISMC Toolkit are shown in Figure 2-1.

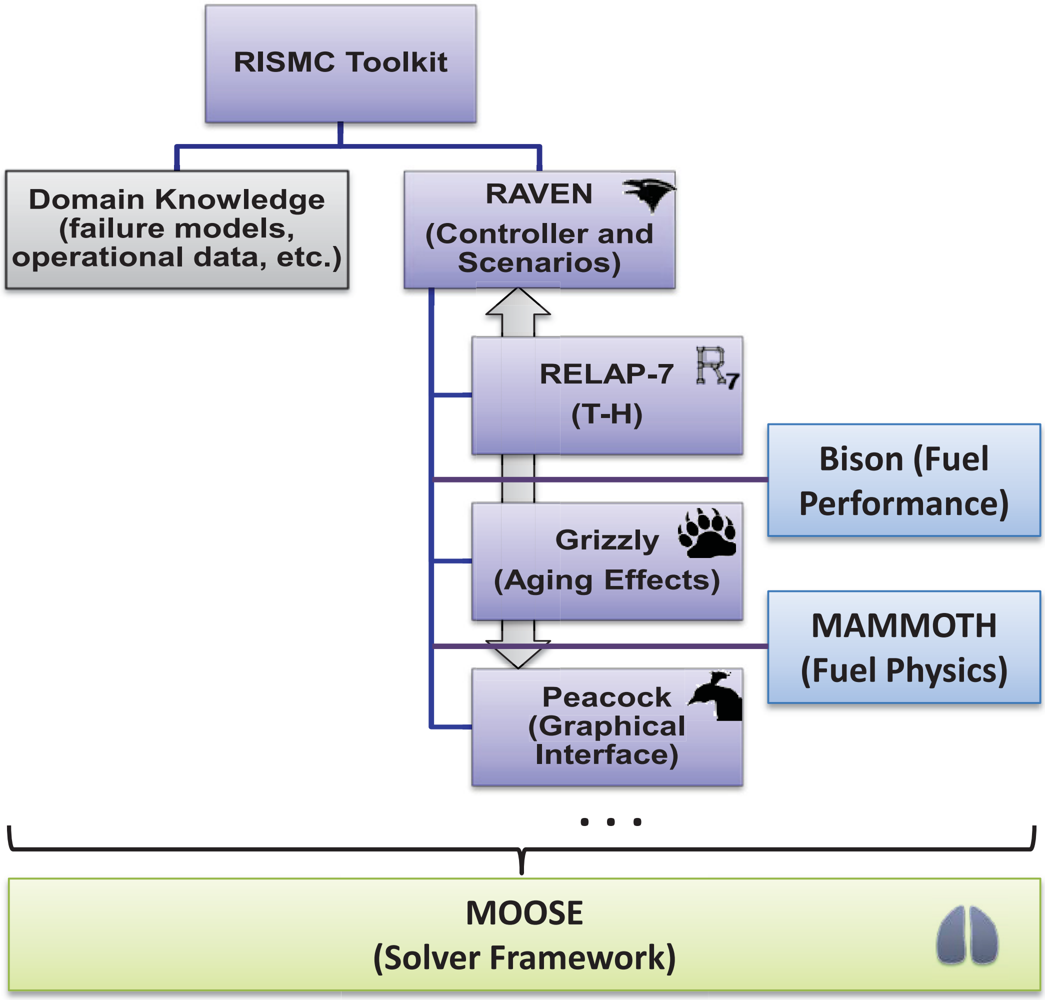

Figure 2-1. The major software modules present in the current RISMC Toolkit (shown coupled to additional modules for fuel performance and fuel physics).

2.2 Details of the R&D Approach

2.2.1 Probabilistic Safety Margin

Central to this Pathway is the concept of a safety margin. In general engineering terms, a "margin" is usually characterized in one of two ways:

- A *deterministic* margin, defined by the ratio (or, alternatively, the difference) of an applied capacity (i.e., strength) to the load. For example, we test a pressure tank to failure where the tank design is rated for a pressure **C**, it is known to fail at pressure **L**, thus the margin is **(L – C)** (safety margin) or **L/C** (safety factor).
- A *probabilistic* margin, defined by the probability that the load exceeds the capacity. For example, we model failure of a pressure tank where the tank design capacity is a distribution f(**C**), its loading condition is a second distribution g(**L**), the probabilistic margin would be represented by the expression Pr[**L** > **C**].

In practice, actual loads (**L**) and capacities (**C**) are uncertain and, as a consequence, most engineering margin evaluations are (or should be) of the probabilistic type (in cases where deterministic margins are evaluated, the analysis is typically conservative in order to account for uncertainties). The RISMC Pathway uses the probability margin approach to quantify impacts to economics, reliability, and safety in order to avoid conservatisms (where possible) and treat uncertainties directly. Further, we use this approach in risk-informed margins management to present results to decision makers as it relates to margin evaluation, management, and recovery strategies.

Probabilistic Safety Margin

A numerical value quantifying the probability that a safety metric (e.g., for an important process variable such as clad temperature) will be exceeded under simulated scenario conditions.

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The types of margins that can be characterized vary according to the "system" of interest and the performance metrics being evaluated. Examples of these metrics are listed in Table 2-1.

Table 2-1. Examples of different types of margins that can be characterized.

"System"	Performance Metric	Example of Margin Contributors
Nuclear Power Plant	Safety margin	L = scenarios are modeled that represent component failures/successes leading to an increased core coolant temperature C = ability of the fuel/clad to withstand elevated core coolant temperature
Structures such as the Core Internals	Economic margin	L = scenarios are modeled that account for potential costs of off-normal conditions and replacement due to core internal degradation issues C = ability of the core internals to withstand radiation embrittlement and corrosion
Component such as an Emergency Diesel Generator	Seismic margin	L = scenarios are modeled that estimate the energy transferred from an earthquake using non-linear soil-structure interaction analysis C = ability of a diesel generator to withstand the energy transferred from the earthquake

As a simplified illustration of the type of approach taken by the RISMC method and tools, we show a hypothetical example in Figure 2-2. For this example, we suppose that a NPP decision-maker has two alternatives to consider: Alternative #1 – retain an existing, but aging, component as-is or Alternative #2 – replace the component with a new one. Using simulation-based risk analysis methods and tools (described in Section 3), we run 30 simulations where this component plays a role in plant response under off-normal conditions. For each of the 30 simulations, we calculate the outcome of a *selected* safety metric – in this example peak clad temperature – and compare that against a capacity limit (assumed to be 2200 F)[a]. However, we have to run these simulations for both alternative cases (resulting in a total of 60 simulations). The results of these simulations are then used to determine the probabilistic margin:

Alternative #1: Pr[L > C] = 0.17

Alternative #2: Pr[L > C] = 0.033

If the safety margin characterization were the *only* decision factor, then Alternative #2 would be preferred (it has a better margin than Alternative #1 since its safety characteristics are better). But, these insights are only part of the decision information that would be available to the decision maker, for example the costs and schedules related to the alternatives would also need to be considered. In many cases, multiple alternatives will be available to the decision maker due to level of redundancy and several barriers for safety present in current NPPs.

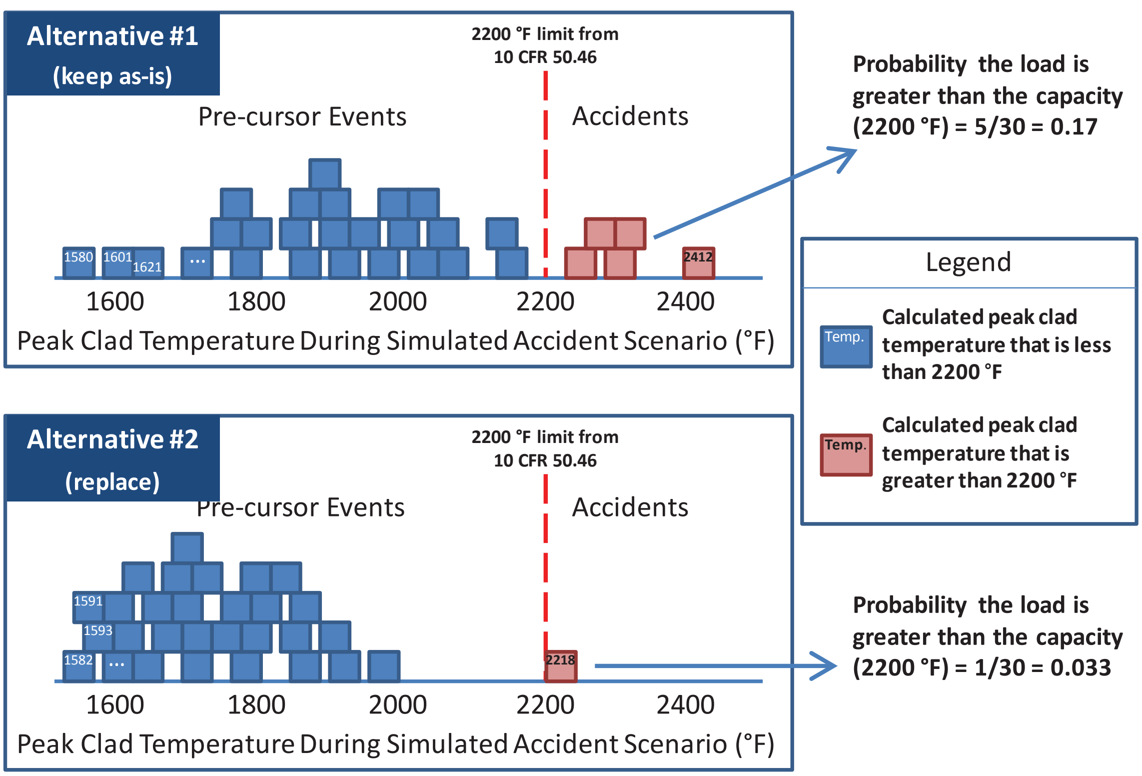

Figure 2-2. RISMC example when evaluating alternatives for risk-informed margins management.

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[a] Note that in this example, the capacity is represented by a single value (2200 F) rather than be a distribution. In general, for a performance metric such as safety, the capacity *would* be represented by a distribution representing the possible variation in the behavior of fuel/clad performance under various plant scenarios and conditions.

If one focuses on a specific scenario shown in Figure 2-2, we can determine the details of the scenario that determined failure or success (i.e., failure is defined as scenarios resulting in a peak cladding temperature in the core greater than 2200 F). Each "box" embodies a single simulation representing a single scenario. This scenario is determined by RAVEN, where we produce scenarios via stochastic simulation. For example, "inside" the first blue box labeled "1580" under Alternative #1, the scenario that is captured could produce the information shown in Figure 2-3.

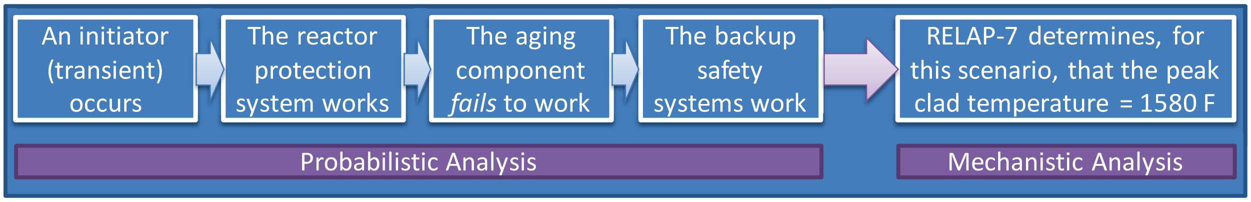

Figure 2-3. Example of the details available from a scenario characterization via simulation.

Because one LWRS Program objective is to develop technologies that can improve the reliability, sustain safety, and extend the life of the current reactors, any safety margin focus would need to consider more realistic load and capacity implications for operating NPPs. For example, the notional diagram shown in Figure 2-4 illustrates that safety, as represented by a load distribution, is a complex function that varies from one type of off-normal scenario to the next. However, the capacity part of the evaluation may not vary as much from one accident to the next because the safety capacity is determined by physical design elements such as fuel/clad and material properties (which are common across a spectrum of off-normal scenarios) or regulatory safety limits (such as the 10 CFR 50.46 limit in the Figure 2-2).

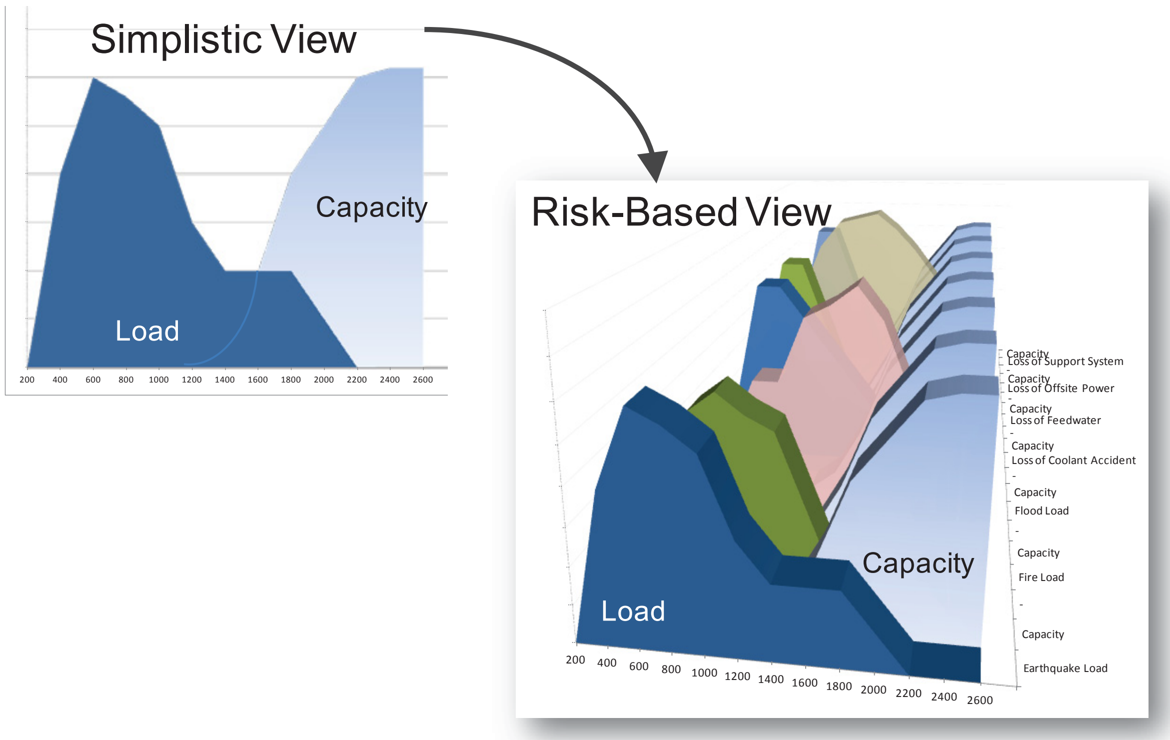

Figure 2-4. Family of load and capacity distributions representing different off-normal conditions.

2.2.2 Safety and Economic Impacts

To successfully accomplish the goals described in Section 2.1, the RISMC Pathway will define and demonstrate the risk-informed safety margin approach. The determination of the degree of a safety margin requires an understanding of risk-based scenarios. Within a scenario, an understanding of plant behavior (i.e., operational rules such as technical specifications, operator behavior, and SSC status) and associated uncertainty will be required to interface with a systems code (i.e., RELAP-7 as part of the RISMC Toolkit). Then, to characterize safety margin for a specific safety performance metric[b] of consideration (e.g., peak clad temperature), the plant simulation will determine time and scenario-dependent outcomes for both the load and capacity. Specifically, the safety margin approach will use the physics-based plant results (the "load") and contrast these to the capacity (for the associated performance metric) to determine if safety margins have been exceeded (or not) for a family of accident scenarios. Engineering insights will be derived based on the scenarios and associated outcomes.

The RISMC Pathway will also develop a significantly improved plant physics code (i.e., RELAP-7) and a suite of control/probability methods contained in RAVEN for driving RELAP-7 to analyze safety margin as part of the RISMC Toolkit. These tools will use advanced computational techniques to simulate the behavior of NPPs in a way that develops more comprehensive safety insights and enables a more useful risk-informed analysis of plant safety margin than can be done using existing tools. RELAP-7 is a systems code, meaning it will simulate behavior at the plant level (i.e., it will address a broad range of

[b] Safety performance metrics may be application-specific, but in general are engineering characteristics of the NPP, for example as defined in 10 CFR 50.36, "safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity."

phenomena at a level of detail that is feasible and appropriate for a plant scale of modeling) as opposed to analyzing highly localized phenomena in great detail at every point in the plant (which is still infeasible today). However, as a systems code, RELAP-7 will function as an environment within which user-supplied, highly detailed models of selected subsystems [e.g., via linkage to the Consortium of Advanced Simulation of Light Water Reactors (CASL) model of the reactor core] can be applied as needed.

The type of "plant physics" represented in the RELAP-7 software will include both T-H and neutronics. Specifically, RELAP-7 has both neutronics physics (via a point kinetics model) and T-H physics (through a variety of models covering single- and two-phase flow). The Pathway is also planning on coupling RELAP-7 to other INL-developed neutronics software (e.g., Rattle$S_N$ake, a S_N transport code being developed using the MOOSE framework) when needed for the spatially-dependent neutronics models. Consequently, for scenarios only requiring "simple" neutronics (which are most accident scenarios) one would just use RELAP-7 and use the built-in neutronics module. More complicated problems (from a neutronics standpoint) would require neutronics from a more sophisticated module, such as the Rattle$S_N$ake transport code [4] or linkage to higher-fidelity models such as those being developed by CASL.

In addition to the safety impacts that are represented in the probabilistic scenarios, the RISMC Pathway is also able to address economic impacts. In the example previously illustrated in Figure 2-2, we considered two alternatives:

Alternative #1 – *retain* the existing, but aging, component as-is

Alternative #2 – *replace* the component with a new one

Each one of these alternatives has an economic impact associated with it. However, the type of costs associated with each is complicated and falls into two general types, direct costs (typically with small uncertainties) and indirect costs (typically with large uncertainties). Examples of these costs are:

- Alternative #1
    - Direct Costs: Inspection or maintenance of the aging component now and in the future.
    - Indirect Costs: The cost associated with pre-cursor events in the future; the cost associated with accidents in the future; the cost to replace the component in the future.
- Alternative #2
    - Direct Costs: The cost to replace the component now.
    - Indirect Costs: The cost associated with pre-cursor events in the future; the cost associated with accidents in the future

For the two alternatives, the direct costs would typically be modeled and quantified by the owner/operators of the specific facility. It is the *other* costs, those that occur probabilistically (i.e., in the future), that is of interest to the RISMC Pathway since our methods and tools can represent and quantify those costs directly as part of the simulation. For example, Figure 2-5 shows, for a specific simulated outcome, how costs would be represented (for both pre-cursor and accident events). Note that even in cases where a peak clad temperature outcome does *not* exceed the 2200F limit, an impact could be that a degraded component can cause an outage (say a pipe ruptures causing damage). In this hypothetical case, the plant would require an outage to repair the damage, which had an economic penalty due to replacement components and lost power generation.

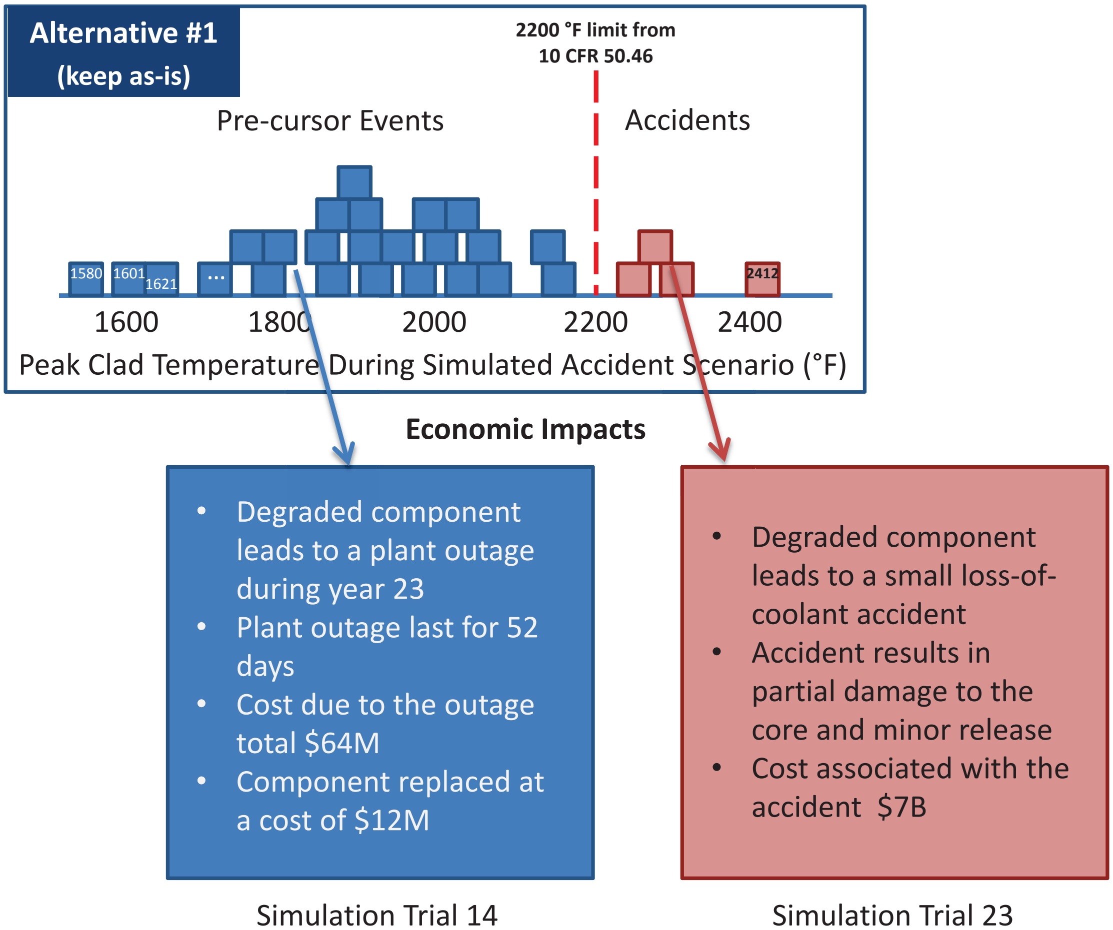

Figure 2-5. Hypothetical example of economic considerations of probabilistic costs as part of risk-informed margins management.

2.3 The Federal Role

In order to better manage the successful operation of the NPP fleet for current- and extended-lifetimes, there are needs to characterize and manage safety margin. Using a risk-informed approach, issues such as what are off-normal conditions, how likely are they, and what are the consequences, are all questions that must be addressed. The RISMC method and associated Toolkit provide answers to NPP operational questions related to safety margins and plant economics. Motivations for DOE involvement in supporting the RISMC Pathway include:

- The need to better characterize and quantify safety margin when considering plant life extension beyond 60 years.

- The need to better integrate data, models, and information from parallel activities such as materials research and instrumentation and controls development. From these complementary activities, we can assimilate potential safety implications in order to better predict NPP viability and to support decision-makers.

- The need to create confidence in a verified and validated approach and tool set that will be applicable in NPP operation and licensing activities. The DOE national laboratory system has broad experience in validation, verification, and uncertainty quantification, which are essential components for successful development of the RISMC Toolkit.

- The need to provide relevant economic information as it pertains to off-normal scenarios, including the incorporation of aging concerns.

- The need to enhance and expand on the existing body of methods and tools. Many of the legacy safety tools in use in the US nuclear power industry were designed and created 30 to 40 years ago.

- The need to better understand beyond design-basis events. As a result of the Fukushima event, NPPs are being asked for information on hazards such as seismic and floods and to characterize the safety impact of these hazards.

- The need to move NPP analysis onto modern high-performance computational architectures, methods, and cloud computing approaches and move away from more-limited techniques.

- The need to use science-based models for prediction of NPP performance rather than parametric- or correlation-based mechanistic models that are prevalent.

- The need to pro-actively respond to future NPP changes over extended life-times (such as aging) or for desired plant changes (such as increasing the economic viability by extended power uprates).

- The need to better describe uncertainties with a focus on improved decision-making.

- The development of tools such as RELAP-7 is high-risk, requiring multi-physics modeling capabilities developed in the DOE national laboratory system. Moreover, RELAP-7 is highly multi-disciplinary, making RELAP-7 development a good match for the institutional conditions at DOE.

- Government and industry are sharing work on methods and tools for characterizing safety margin.

    - The DOE role is to lead the development of advanced techniques, including building on uncertainty analysis methodology that has been under development for years at government laboratories and internationally.

    - Industry, under EPRI's Long-Term Operation Program, is carrying out case studies to better understand the issues and to provide feedback and comparative results to DOE on both RELAP-7 development and the methods and tools for analysis of safety margin.

One result of the approach in the RISMC Pathway is the use of risk informed margins management strategies. These strategies will be informed by the risk and economics assessment and will focus on desired, measurable outcomes, rather than prescriptive processes, techniques, or procedures, with the aim of identifying performance measures that ensure an adequate safety margin is maintained over the lifecycle of a NPP. In addition to the activities identified above in this pathway, RISMC will be working with the Materials and the Advanced Instrumentation and Control Systems Technologies Pathways. In addition, the RISMC Pathway will be collaborating with the DOE Advanced Fuels Campaign on risk informed case studies for issues such as accident tolerant fuel design and testing.

3. RISMC RESEARCH AND DEVELOPMENT AREAS

The purpose of this section is to describe those R&D areas that are the focus for the RISMC Pathway.

To better understand the approach to determine safety margins, we first describe the two types of analysis used in this pathway (see Figure 3-1), probabilistic and mechanistic quantification. Note that in actual applications, a blended approach is used where both types of analysis are combined to support any one particular decision.

Types of Analysis Used in Safety Margin Evaluations	
PROBABILISTIC	MECHANISTIC
Pertaining to stochastic (non-deterministic) events, the outcome of which is described by a probability.	Pertaining to deterministically predictable events, the outcome of which is known with certainty if the inputs are known with certainty.
Probabilistic analysis uses models representing the randomness in the outcome of a process. Probabilities are not observable quantities, we rely on models to estimate them for certain specified outcomes such a failure of a component.	Mechanistic analysis (also called "deterministic") uses models to represents situations where the observable outcome will be known given a certain set of parameter values.
An example of a probabilistic model is related to counting of *j* number of failures of an operating component in time *t*: Probability($j>0$) = $1-e^{-\lambda t}$.	An example of a mechanistic model is the one-dimensional transfer of heat (or heat flux) through a solid: $q = -k\, \partial T/\partial x$.

Figure 3-1. Types of analysis that are used in the RISMC Pathway.

The use of both types of analysis, probabilistic and mechanistic, is represented in Figure 3-2. Probabilistic analysis is represented by the risk analysis while mechanistic analysis is represented by the plant physics calculations. Safety margin and uncertainty quantification rely on plant physics (e.g., thermal-hydraulics and reactor kinetics) coupled with probabilistic risk simulation. The coupling takes place through the interchange of physical parameters (e.g., pressures and temperatures) and operational or accident scenarios. Together, the analysis methods can be used to support a variety of safety margin decisions, including recovery of or increasing safety margins:

- If the nominal core power levels are increased (power uprate)
- If a different type of fuel or clad is introduced
- If aging phenomena becomes more active over long periods of plant operation
- If plant modifications are taken to increase resiliency for hazards such as flooding and seismic events
- If systems, structures, or components are degraded or failed

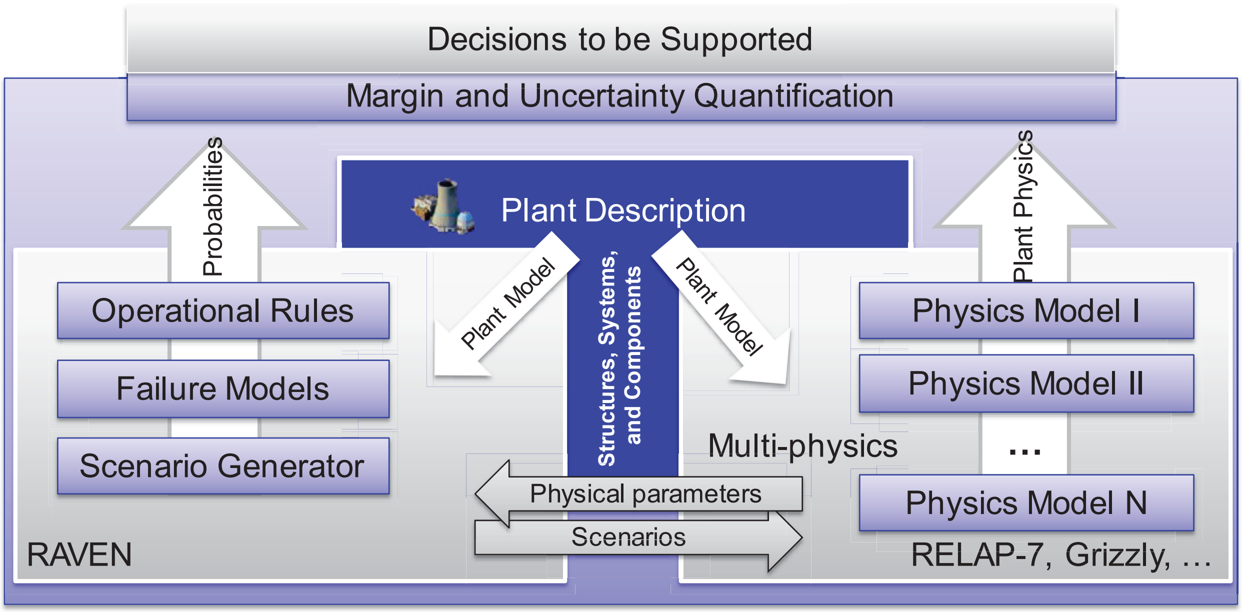

Figure 3-2. Attributes of the RISMC approach for supporting decision-making.

The RISMC Pathway has two primary focus areas to guide the R&D activities.

1. The Pathway is developing the technical basis of methods for safety margins quantification in support of the risk-informed decision making process.

2. The Pathway is producing an advanced set of software tools used to quantify safety margins. This set of tools, collectively known as the RISMC Toolkit, will enable a risk analysis capability that currently does not exist.

3.1 Technical Basis for Risk-Informed Margins Management

The RISMC methodologies are captured in a set of technical basis reports. These guides are technical documents that describe how the RISMC Pathway captures the protocols for analysis and evaluation related to safety margin characterization. The technical basis reports are intended to be companion documents to EPRI-produced reports. The guides will be developed to support industry use in their Risk-Informed Margins Management (RIMM), plant analysis, and licensing activities.

3.1.1 The Safety Case

The technical basis for risk-informed margins management is captured in what is known as the "safety case." While definitions may vary in detail, the safety case essentially means the following:

> *A structured argument, supported by a body of evidence that provides a compelling, comprehensible and valid case that a system is adequately safe for a given application in a given environment. [5]*

The realization of a safety case for RISMC applications will be an output when applying the Pathway methods. The safety-margin claims will do the following:

1. Make an explicit set of safety margin claims about the facility and its constituent SSCs.

2. Produce qualitative and quantitative evidence that supports the claims from #1.

3. Provide a set of safety margin management strategies that link the claims to the probabilistic and mechanistic evidence.

4. Make clear the assumptions, models, data, and uncertainties underlying the arguments

5. Allow different viewpoints and levels of detail in a graded fashion to support decision making.

The safety case of a facility or particular SSCs should be regarded as having fundamental significance as opposed to being mere documentation of facility or SSC features. For practical purposes, "safety margin" is not observable in the way that many other operational attributes are (e.g., core temperature or embrittlement of pressure vessels). However, in decision-making regarding the facility or SSC margin management strategies, the safety case is a proxy for a set of safety attributes of interest. And, regardless of context, the formulation of a safety case is about developing a body of evidence and marshaling that evidence to inform a decision.

Since safety margins are inferred (not directly observable) unlike how cost, power output, pipe thickness, water temperature, radiation level, etc., are observed, we rely on a combination of models (probabilistic and mechanistic) to make safety margin predictions. These models also rely on unobserved elements such as failure rates and probabilities. Consequently, the characterization of a safety margin requires the treatment and understanding of uncertainty in order to effectively manage margins in a risk-informed decision making approach. Further, the decision of what is adequate margin resides with the NPP decision-makers and is informed by our models, sensitivity cases using those models, and other information in an integrated approach.

3.1.2 Margins Analysis Techniques

One aspect of the technical basis that is addressed is the mechanics of techniques to conduct margins analysis, including a methodology for carrying out simulation-based studies of safety margins, using the following process steps (as shown in Figure 3-3) for RISMC applications.

1. Characterize the issue to be resolved in a way that explicitly scopes the modeling and analysis to be performed, including delineating the performance metrics to be analyzed (e.g., safety, economics).

2. Quantify the relevant state-of-knowledge (i.e., uncertainty) of the key variables and models for the issue at hand. For example, describe parametric uncertainties to be sampled during the analysis in later steps.

3. Determine issue-specific, risk-based scenarios and associated timelines (as depicted in Figure 3-4). The scenario simulation captures timing considerations that may affect plant physical phenomena and margins, as described in Steps 4 and 5. As such, there will be strong interactions between the analysis Steps 3-5. Also, in order to "build up" the load and capacity distributions representing the safety margin (as part of Step 6), a large number of scenarios will be needed to be evaluated.

4. Represent plant operation probabilistically using the scenarios identified in Step 3. For example, plant operational rules (e.g., operator procedures, where we include the possibility for human-caused failures) are used to provide realism for scenario generation. Because numerous scenarios will be generated, the plant- and operator-behavior cannot be manually created like in current risk assessment using event- and fault-trees. In addition to the *expected* operator behavior, the probabilistic plant representation will account for the possibility of failures.

5. Represent plant physics mechanistically. The plant systems level code (e.g., RELAP-7) will be used to develop time and/or space distributions for the key plant process variables (i.e., loads). Other codes such as fuel/clad performance codes will be used to develop capacity distributions. Because there is a coupling between Steps 4 and 5, they each can impact the other. For example, a calculated high loading (from pressure, temperature, or radiation) in an SSC may disable a component, thereby impacting an accident scenario that challenges fuel performance.

6. Construct and quantify the load and capacity distributions (obtained from Steps 4 and 5) relating to the performance metrics that will be analyzed to determine the margin.

7. Determine how to manage *uncharacterized* risk. Because there is no way to guarantee that all scenarios, hazards, failures, or physics are addressed, the decision maker should be made aware of limitations and uncertainties in the analysis. This step relies on effective communication from the analyst in order to understand the risks that *were* characterized.

8. Identify and characterize the items that determine the relevant margins within the issue being evaluated to in order to develop appropriate RIMM strategies. Determine whether additional work to reduce uncertainty would be worthwhile or if additional (or relaxed) controls are justified.

One of the unique aspects of the RISMC approach compared to traditional PRA is how it couples probabilistic approaches (the scenario) directly with mechanistic phenomena representation (the physics) through simulation. This simulation-based modeling allows decision makers to focus on one or more safety, performance, or economic metrics. For example, while traditional risk assessment approaches attempt to quantify core damage frequency (CDF), RIMM approaches may instead wish to consider other metrics such as:

- Magnitude of the hazard – for example, when evaluating external hazards, the height of water on buildings, or the height of water inside strategic rooms. The "magnitude" might be measured (during the simulation) by metrics such as water height, seismic energy, water volume, water pressure, etc.
- Damage to the plant (but not core damage) – for example, we may be interested in scenarios in which the facility does not see core damage, but would still experience extensive (or even minor) damage. The "damage" might be measured (again during the simulation) by metrics such as total number of components failed, cost of components destroyed, structures rendered unusable, the length of time the facility is impacted (hours versus months), etc.

The defining difference between these new RIMM metrics and traditional ones such as CDF is that they represent observable quantities (e.g., the number of components failed, the costs related to the event, the height of water in a room, the duration of the event) rather than just a statistical average of an event frequency. We believe these new metrics that are provided by the RISMC simulation yield enhanced decision-making capabilities for nuclear power plants.

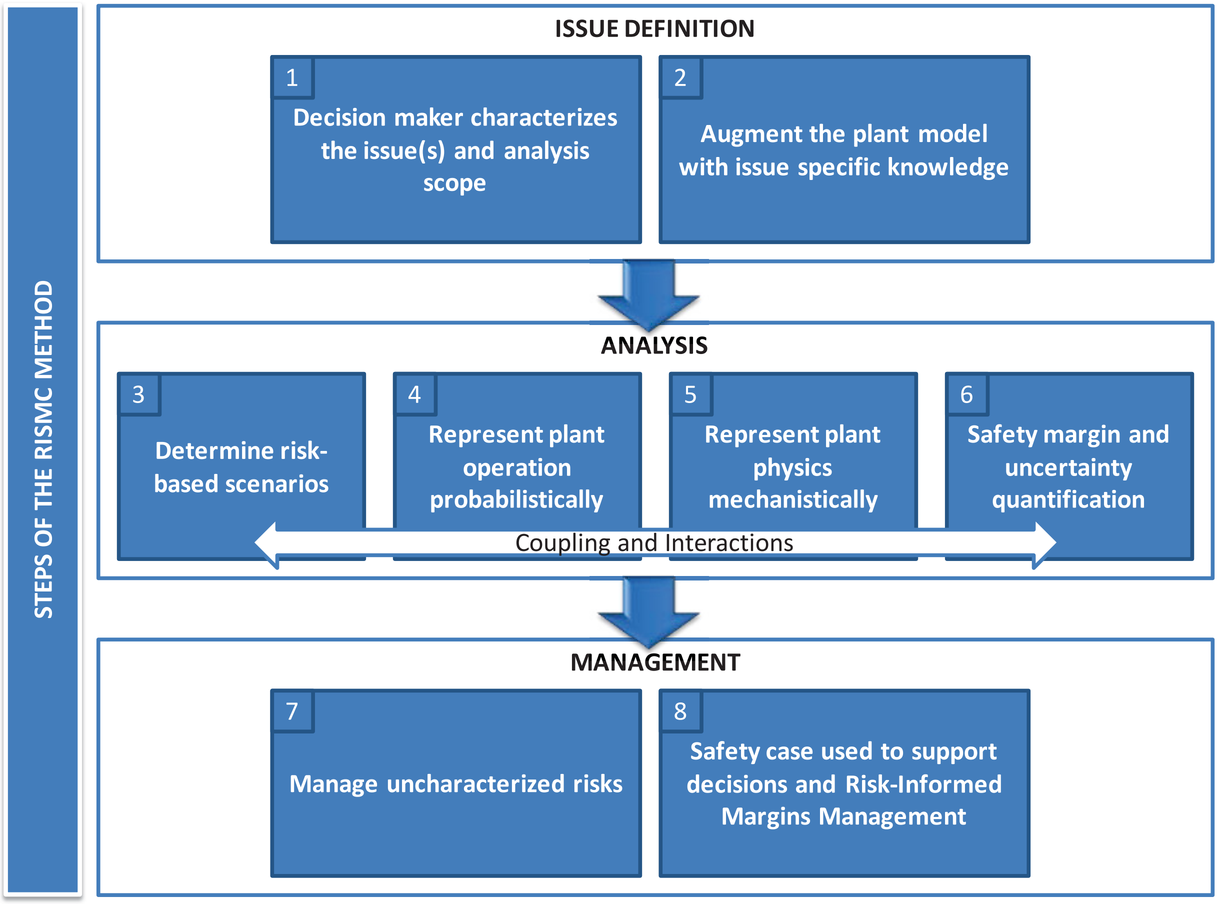

Figure 3-3. Depiction of the high-level steps required in the RISMC method.

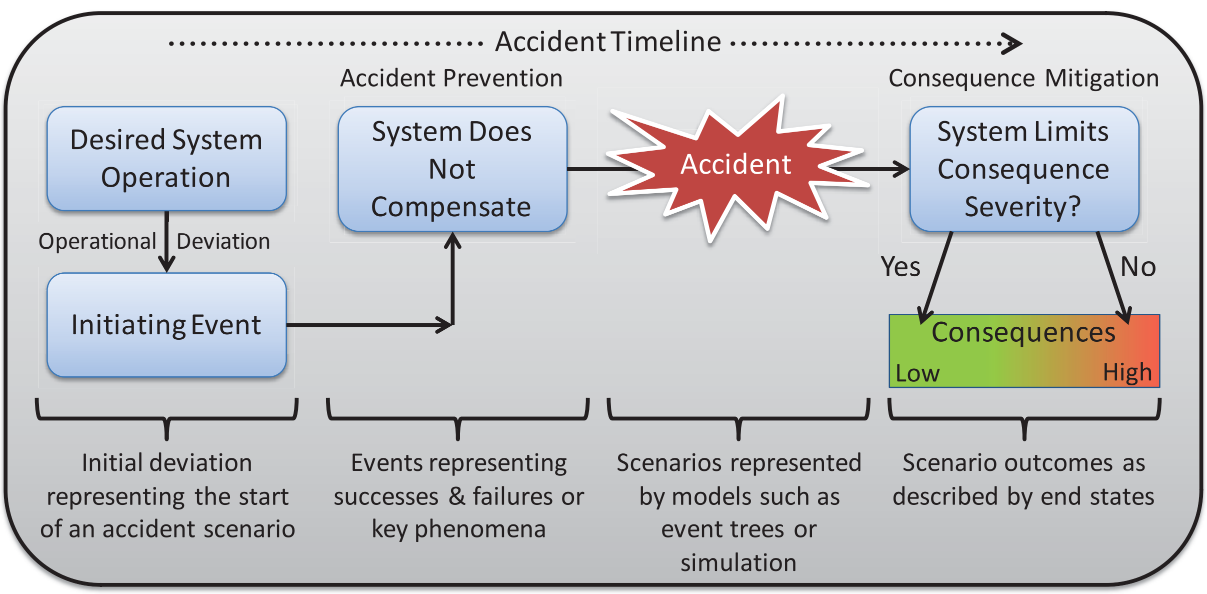

Figure 3-4. Accident scenario representation.

3.1.3 Case Study Collaborations

Jointly with EPRI, the LWRS RISMC Pathway is working on specific case studies of interest to the NPP industry. During FY2013 through FY2015, the team performed multiple case studies including a demonstration using the INL's Advanced Test Reactor, a hypothetical pressurized water reactor, and a boiling water reactor extended power uprate case study. Safety margin recovery strategies will be determined that will mitigate the potential safety impacts due to the postulated increase in nominal reactor power that would result from the extended power uprate. An additional task was to develop a technical report that describes how to perform safety margin-based configuration risk management. Configuration risk management currently involves activities such as the Significance Determination Process which traditionally uses core damage frequency as the primary safety metric – the research will focus on how the safety-margin approach may be used to determine risk levels as different plant configurations are considered. The results for the recent case studies are briefly described here.

3.1.3.1 Advanced Test Reactor Case Study

Constructed in 1967, ATR is a pressurized water test reactor that operates at low pressure and low temperature. It is located at the Advanced Test Reactor Complex on the INL site. The reactor is pressurized and is cooled with water. The reactor vessel is a 12-ft diameter cylinder, 36-ft high, and is made of stainless steel. The reactor core is 4 ft in diameter and height and includes 40 fuel elements capable of producing a maximum power of 250 MW. The reactor inlet temperature is 125°F and the outlet temperature is 160ºF. The reactor pressure is 390 pounds per square inch.

As part of the RISMC demonstration, we successfully coupled the risk assessment simulation to the thermal-hydraulics analysis in order to integrate probabilistic elements with mechanistic calculations. With the knowledge of plant response, we needed to determine whether or not a particular outcome is "success" (meaning no fuel damage) or "failure" (meaning fuel damage). For our analysis, we assumed that any event that saw a peak cladding temperature of 725°F (658 K) was a fuel damage outcome.

The purpose of the RISMC ATR case study was to demonstrate the RISMC approach using realistic plant information, including both real PRA and thermal-hydraulics models. As part of this case study, we evaluated emergency diesel generator issues. Historically, ATR has had a continually running emergency diesel generator as a backup power supply, which is different than all commercial nuclear power plants in the United States (commercial plants have their emergency diesel generators in standby). Margin recovery strategies under consideration include the following:

- Keep the emergency power system as is (emergency diesel generator running, one in standby, and commercial power as backup)
- Redundant commercial power as primary backup, single new emergency diesel generator as backup
- Redundant commercial power as primary backup, two existing emergency diesel generators as backup.

For the different strategies, we simulate the plant behavior both probabilistically and mechanistically. To perform this simulation, we used the existing PRA and thermal-hydraulics information. We then defined the simulation for different scenarios and different strategies and ran a large number of iterations to determine overall safety margins. The results vary for each alternative (the margins are different), but are used to determine preferential strategies.

3.1.3.2 Boiling Water Reactor Station Blackout

The scope of the boiling water reactor (BWR) station blackout (SBO) case study is to show the RISMC capabilities in order to assess performances of the power uprates using a simulation-based environment. Such assessment cannot be naturally performed in a classical PRA-based environment since the thermal-hydraulics (T-H) is not integrated with the probabilistic modeling. In our analysis, we used RELAP T-H software and RAVEN as tools to perform a simulation-based stochastic analysis. [6] [7]

The focus of the analysis was to investigate the (possible) impact of power uprate on the safety margin of a BWR. The case study considered is a loss of off-site power followed by the possible loss of all diesel generators, i.e., a station black-out (SBO) event. We created the necessary inputs file for the mechanistic T-H codes that models system dynamics under SBO conditions. We also interfaced RAVEN with these codes so that it would be possible to run multiple RELAP simulation runs by changing specific portions of the input files. We employed classical statistical tools, i.e. Monte-Carlo, and more advanced machine learning based algorithms to perform uncertainty quantification in order to determine changes in system performance and limitations as a consequence of power uprate. We also employed advanced data analysis and visualization tools that helped us to correlate simulation outcomes such as maximum core temperature with a set of input uncertain parameters.

Results obtained give a detailed overview of the issues associated to power uprate for a SBO accident scenario. For example, we were able to quantify how the timing of specific events was impacted by a higher reactor core power. Such safety insights can provide useful information to the decision makers to perform risk-informed margins management.

As an example of the RIMM insights gained from the RISMC analysis, Figure 3-5 shows the limit surface for two different core power levels where variations in either off-site (i.e., AC power) recovery time or the time at which the diesel generators fail (i.e., DG failure time) can affect the outcome of core damage (failure) or not (success). As can be seen in the figure, as the core power level is increased, it becomes more likely to see core damage. In the nominal case, if off-site power is recovered in less than 7 hours (approximately 25,000 seconds) then core damage is always averted. However, in the 120% power uprate case, in scenarios where the diesel generators fail early (in less than 2 hours) and off-site power is recovered in less than 7 hours, some of those cases result in core damage.

Limit Surface

The boundary in the input space between two simulated outcomes, for example, failure or success.

Limit Surface

The boundary in the input space between two simulated outcomes, for example, failure or success.

Further, the limit surface is determined by understanding the outcomes of specific scenarios (shown as the red and green points in Figure 3-5). These individual points (or scenarios) provide information to the decision maker for use in RIMM. For example, in the case of the power uprate to 120% power, the new points/scenarios that result in core damage may be investigated to determine what at the plant may be changed in order to mitigate the core damage risk while still maintaining an economic enhancement of the power uprate.

Note that in order to characterize the limit surface, the concept of "margin" has already been resolved. In order to define a specific point around the limit surface, we had to determine, during the simulation, if the load exceeded the capacity for the performance metric of interest (e.g., safety). For cases where the load exceeded the capacity, we classify these scenarios as a "failure" (the red points). Thus, over a large number of simulation cases, we are able to quantify the margin probabilistically by determining the expression Pr[L > C]. One interesting fact is that the limit surface itself is the location where the margin is equal to zero – it represents cases where the load equals the capacity.

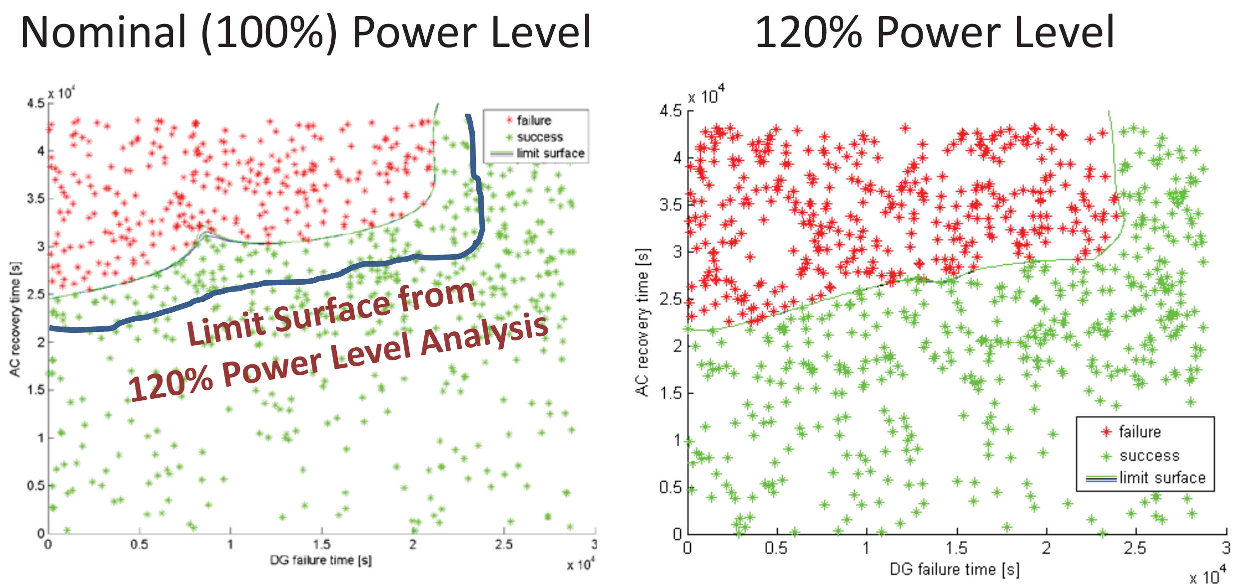

Figure 3-5. Limit surface plots from the BWR SBO analysis.

3.1.3.3 Safety Margin Configuration Risk Management

Configuration risk management is an important process that evaluates testing and maintenance activities that are proposed while the reactor is at full power. Performing these activities while at full power provide many benefits to the plant owner/operators, improving both economics and safety. Configuration risk management helps identify if these activities should be allowed while at power based on their risk impact. The proposed configuration is evaluated and if the increase in the risk metric of choice does not exceed a predefined safety threshold then the planned activities can proceed. Different plant configurations, depending upon safety system and duration, can have different impacts on risk.

Configuration risk management is also used to evaluate degraded conditions that have occurred at plants. This evaluation is based on the plant configuration during the degraded condition to assess what the increase in risk was observed. Given this information, management changes can be implemented to decrease the future likelihood of being in these degraded conditions.

R&D on configuration risk management using the RISMC approach showed how improved accuracy and realism can be achieved by simulating changes in risk – as a function of different configurations – in order to determine safety margins as the plant is modified. [8] In order to carry out configuration risk management, a coupling of mechanistic and probabilistic calculations is performed. Within this process, several technical issues are encountered and addressed so that future applications can take advantage of the analysis benefits while avoiding the technical pitfalls that are found for these types of calculations. The technical areas that were addressed: common cause failure treatment, human error probability determination, incorporation of plant physics, how to perform delta risk calculations, accuracy related to convolution factors, and resolving success states as part of the modeling. For each technical issue, specific recommendations were provided with the intention of improving the safety margin analysis and strengthening the technical basis behind the analysis process. By following the overall RISMC approach described and applying the recommendations, a technically-sound safety margin characterization for configuration risk management can be realized.

3.1.3.4 External Events Analysis

In FY14, the RISMC Pathway extended its analysis capabilities into additional initiating events including external events (primarily focusing on seismic and flooding events). The approach used to treat an event such as flooding is illustrated in Figure 3-6 and follows:

1. *Initiating event modeling*: modeling characteristic parameters and associated probabilistic distributions of the event considered

2. *Plant response modeling*: modeling of the plant system dynamics

3. *Components failure modeling*: modeling of specific components/systems that may stochastically change status (e.g., fail to performs specific actions) due to the initiating event or other external/internal causes

4. *Scenario simulation*: when all modeling aspects are complete, (see previous steps) a set of simulations can be run by stochastically sampling the set of uncertain parameters.

5. Given the simulation runs generated in Step 4, a set of statistical information (e.g., margin, core damage probability) can be generated. We are also interested in determining the limit surface.

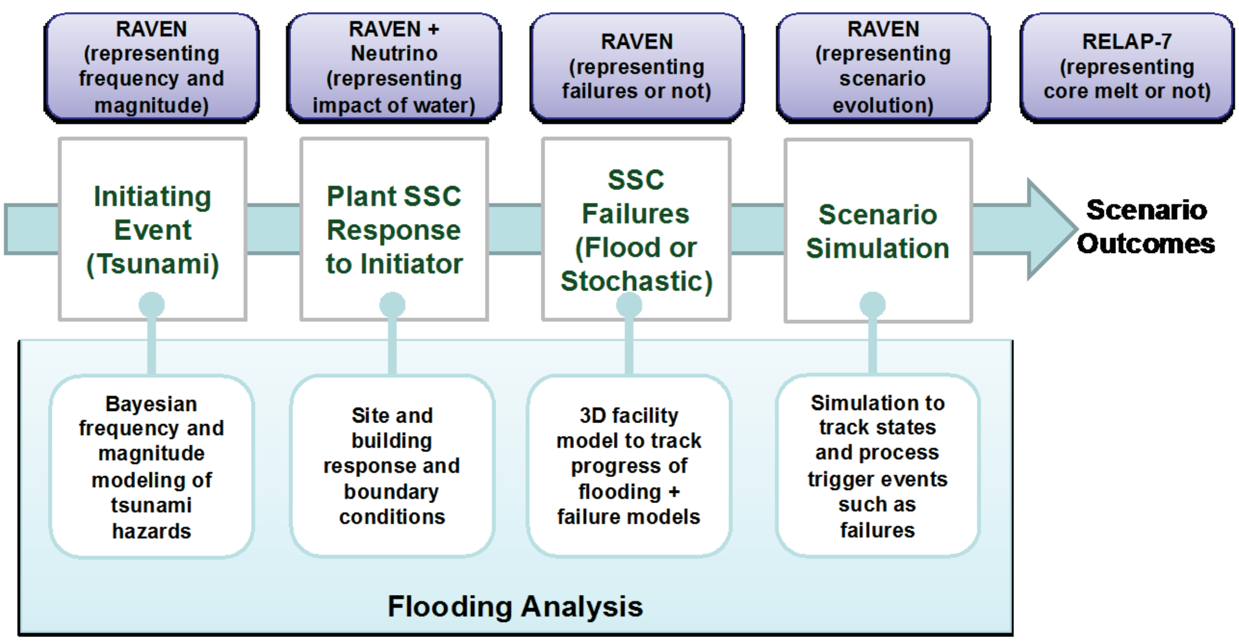

Figure 3-6: Overview of the RISMC scheme to simulate initiating event and plant response using the RISMC toolkit

To demonstrate the RISMC approach for flooding, a generic 3D facility model (see Figure 3-7) with conditions similar to the Fukushima incident was created and used to simulate various tsunami flooding examples. For initial testing only, a slice of the entire facility (containing just a single unit) was used, this includes:

- Turbine building
- Reactor building
- Offsite power facilities and switchyard
- Diesel generator (DG) building

The 3D model is used as the collision geometry for any simulations. For the initial demonstration all objects are fixed rigid bodies – future analysis will explore the possibility of moving debris (caused by the flood) and possible secondary impacts due to this debris.

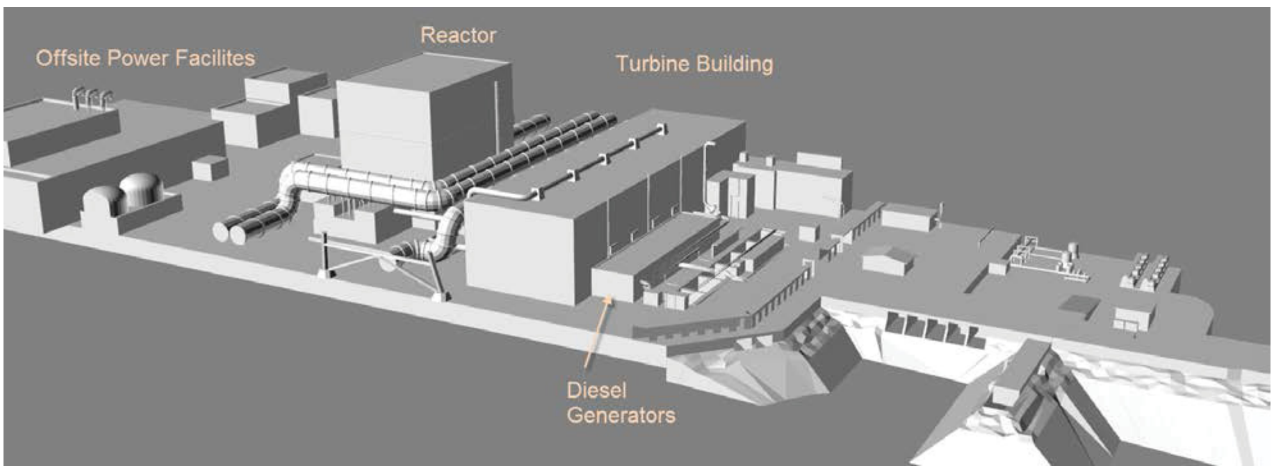

Figure 3-7: 3D plant model developed to simulate flooding

To mimic a tsunami entering the facility, a bounding container was added around the perimeter of the model and for the ocean floor. Then, for this demonstration, over 12 million simulated fluid particles were added for the ocean volume. A wave simulator mechanism was constructed by having a flat planar surface that moves forward and rotates, pushing the water and creating a wave in the fluid particles. Once the wave is "started," the fluid solver handles all of the remaining physics calculations in order to simulate the moving wave through the facility.

As the particles of a simulation move, they interact with the rigid bodies of the 3D model. The simulated fluid flows around buildings, splashes, and interacts in a similar manner to water. Measuring tools can also be added to the simulation to determine fluid contact information, water height, and even flow rates into openings at any given time in the simulation. This dynamic information can be used in two ways: (1) a static success or failure of components or structures depending on wave height or (2) a dynamic result based on time for use in more detailed analysis. As shown in Figure 3-8, the fluid particles are penetrating both air intake vents for an 18 m wave. Evaluating this scenario in more detail, we can determine that at simulation time (or frame) 1,275 DG1 fails from splash particles and DG2 fails at 1,375. Additional detail on the flooding analysis can be found in [9].

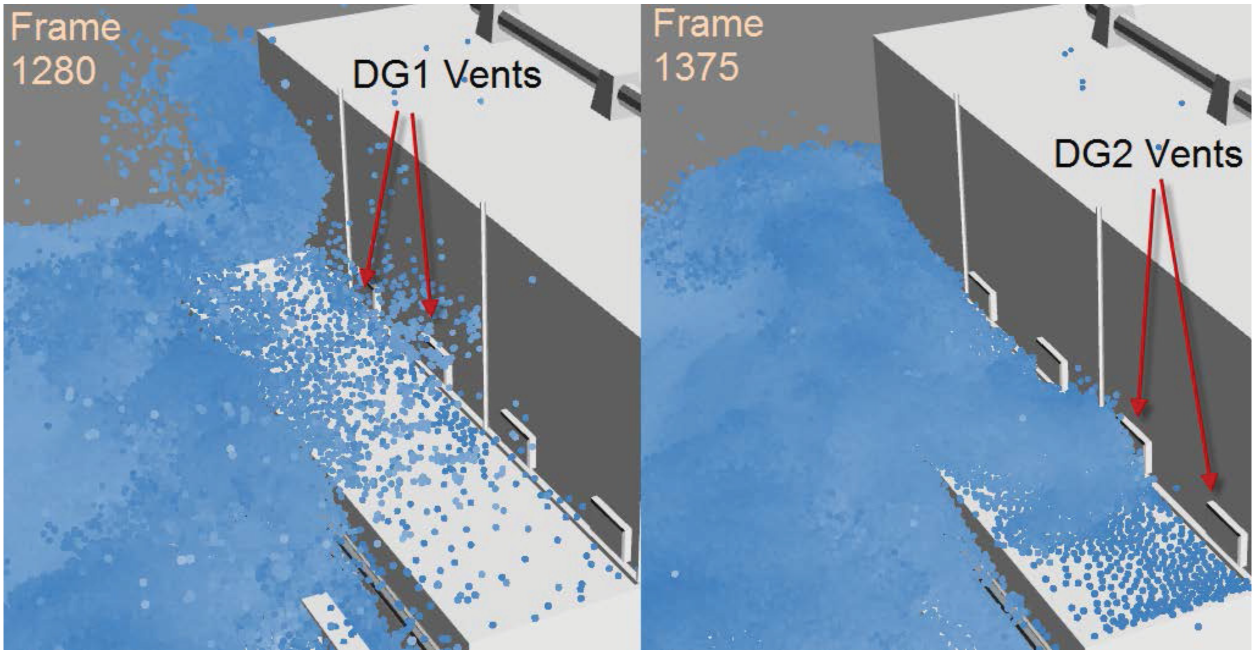

Figure 3-8: Time spacing between failures of generators due to fluid in the air intake vents of the generator room.

In addition to the flooding research, the RISMC Pathway is also investigating advanced methods, tools, and data for seismic hazards. Currently, the nuclear industry is addressing the Fukushima Near-Term Task Force recommendations. One specific recommendation that is of interest is recommendation 2.1 which states, "Order licensees to reevaluate the seismic and flooding hazards at their sites against current NRC requirements and guidance, and if necessary, update the design basis and SSCs important to safety to protect against the updated hazards." On February 15, 2013 the NRC provided its endorsement of the EPRI-1025287 document that serves as a response to the recommendation 2.1. This document provides a process to meet the recommendation 2.1 and includes a screening process that evaluates updated site-specific seismic hazard. One of the items is to evaluate updated site-specific hazard curves, which have the potential for higher magnitude and higher frequency content accelerations.

The NRC requires ten sites to submit their detailed risk analysis by June 30, 2017, while an additional ten sites to submit their detailed risk analysis by December 31, 2019. They are also considering detailed submittals for another 23 sites (by 2020 if needed). And there may be additional requirements "in the near future" related to evaluations of spent fuel pools at some sites. These activities over the next five years are related to the seismic hazard reevaluation and detailed risk evaluation initiatives. In addition to these initiatives, the Near Term Task Force recommendation 2.2 requires plants to perform an update of seismic modeling every 10 years. As part of RISMC, we are bringing a risk-informed picture to seismic analysis by performing evaluations in an integrated fashion, for example through our seismic and flooding work. We are also looking to improve our methods, tools, and data for seismic simulation in order to reduce conservatisms (where they exist) and make the analysis process itself more efficient. Our research and development on seismic analysis will not only improve the safety state-of-the-practice related to seismic analysis, it is timely (by fitting well within the NRC's 2020 time window and 10 year update cycle).

Seismic PRAs are intended to provide best estimates of the various combinations of structural and equipment failures that can lead to a seismic-induced core damage event and the integration of these results to quantify the risk. The advanced seismic PRA methods, tools, and data for RISMC propose to increase the fidelity of the seismic PRA methodology by using high fidelity modeling and simulation tools to provide enhanced seismic calculations for given earthquake events. This advancement is important since NPP evaluations may find that the traditional conservative approach to seismic PRA might produce core damage frequency numbers that are above the NRC allowable limit.

The earthquakes that have been seen recently seem to indicate that traditional seismic models might be conservative. Three recent earthquakes (Kashiwazaki-Kariwa in 207, Fukushima in 2011, and North Anna in 2011) have demonstrated events that exceeded the plant design basis earthquake values. Yet for all of these events, it appears that little damage actually occurred to safety (and most non-safety) related components or structures. By reevaluating these events using a modern analysis approach, we have the opportunity to determine if conservatisms exists in traditional seismic PRA and, if possible, how these conservatisms might be reduced.

3.1.3.5 Collaborations with other LWRS Pathways

In addition to the case study collaborations, the RISMC Pathway works with the other LWRS Pathways, including:

- Collaboration with the Materials Aging and Degradation Pathway on incorporating the insights gained through the Pathway R&D, focusing on material degradation modeling. The primary interaction is through the joint development of the Grizzly tool.

- Collaboration with the Advanced Instrumentation, Information, and Control Systems Technology Pathway on human reliability modeling and non-destructive research on concrete aging predictive modeling.

- Collaboration with the Advanced Reactor Safety Technologies Pathway on leveraging the RISMC Toolkit for use on safety analysis techniques.

3.1.4 Methodology Research Impact

Already, the RISMC R&D is having impacts that are being seen in the industry:

- At the 2012 American Nuclear Society Winter Meeting (in San Diego), one of the RISMC researchers (Diego Mandelli) presented a paper co-authored with Curtis Smith describing the implementation of adaptive sampling algorithms to identify boundaries between system failure and system success. Following the Winter Meeting, Dr. Mandelli and Dr. Smith were recognized for the publication of this paper with an Honorable Mention Award by the Nuclear Installations Safety Division. The paper described an artificial-intelligence based algorithm that is able to drastically reduce the number of simulation needed in order to identify boundaries between important system characteristics such as failure/success. The paper presented both the mathematical background and test cases. In addition, it also demonstrated the algorithm validity for a station blackout analysis. Such sampling analysis allowed the state of the test system to be readily identified (thereby speeding up calculations) when parameters of interest are varied as part of the scenario simulation.

- The RISMC team had a second award winning paper at the American Nuclear Society Probabilistic Risk Assessment 2013 conference. The paper was titled "Adaptive Sampling Algorithms for Probabilistic Risk Assessment of Nuclear Simulations" and was coauthored by Diego Mandelli of the INL.

- The RISMC team had a third award winning paper; a paper written by the RISMC team has been selected among the top 40 papers at the NURETH conference (held in Chicago in September, 2015) as a candidate for best paper. Since there were almost 800 papers at the conference, this places the paper in the top 5% of papers at the conference. The paper was titled "10 CFR 50.46c Rulemaking: A Novel Approach in Restating the LOCA Problem for PWRs" and was authored by Cesare Frepoli, Joseph Yurko, Ronaldo Szilard, Curtis Smith, and Robert Youngblood.

In addition to the award-winning papers, publications in nuclear industry trade journals have been realized, including feature articles in both the ANS Nuclear News and Nuclear Engineering International.

3.1.5 Experimentation

One of the technical gaps that have been identified related to the external events analysis is the lack of fragility models, especially in the area of flooding analysis. Historically, rudimentary conservative methods for conducting flooding risk assessment fall short because they fail to sufficiently characterize both the fragility of components and the risk from the hazards. In short, current risk analysis methodology assumes that many components simply fail if contacted by water. As part of the RISMC Pathway, we propose to test representative nuclear power plant components and structures to failure (and, potentially, to the point of recovering after failure) and develop a science-based approach to flooding risk analysis. We will conduct wave impact, rising water, and top-down water spray testing as part of this experimentation for both mechanical and electrical components. The experimental laboratory will also integrate flood simulation computer codes with the experimental work in order to conduct modeling-informed experimental design. The experimental data obtained will ultimately be used in conjunction with the simulations to create more accurate flooding risk assessment. An example of the type of failure information that will be collected as part of the RISMC flooding experimentation is shown in Figure 3-9. This type of fragility information is needed for flooding in order to have failure models that are equivalent (approximately) in scope and fidelity as those found for other failure modes (e.g., seismic analysis, “random” failures).

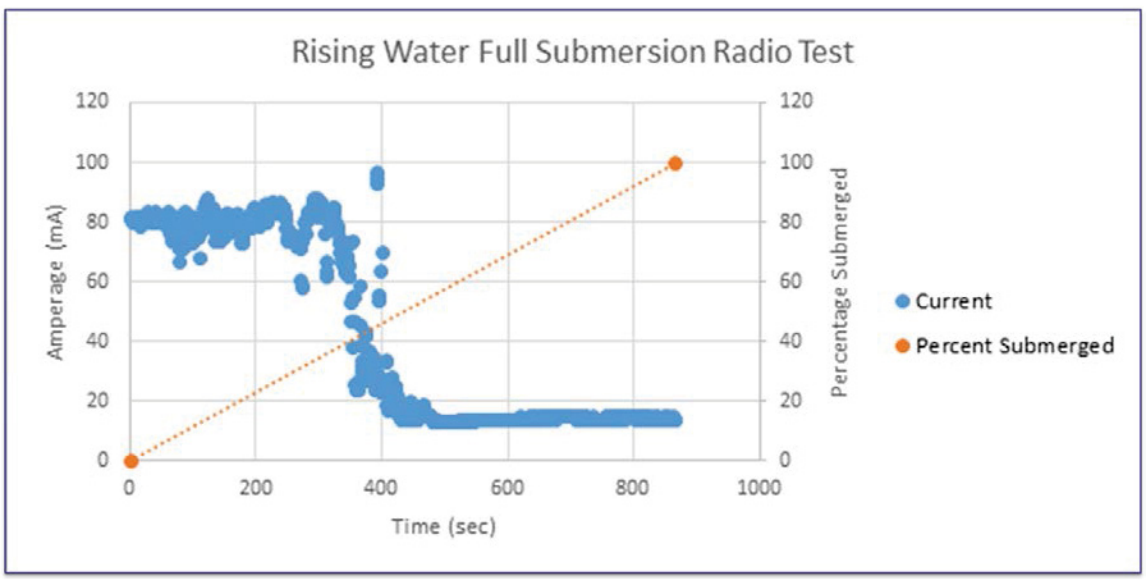

Figure 3-9: Operability, as measured by current draw, of a radio as a function of water inundation.

3.1.6 Validation

Verification, validation, and uncertainty quantification is essential to producing tools that can (and will) be used by industry. Evaluation of existing data for validation is done in parallel with RISMC Toolkit development; verification is done as part of the MOOSE development process. If additional data are needed, experiments will be designed and carried out to meet the validation needs. Tools for uncertainty quantification that can be used with MOOSE-based tools are under development in DOE Programs such as NEAMS and internally at INL, and will be used with the RISMC toolkit. As the development and capabilities of the RISMC Toolkit progress, the LWRS Program will work with industry to determine how to transition the tools to a user-supported community of practice, including planning for lifecycle software management issues such as training, software quality assurance, and development support. The general approach to toolkit development is that the tools will be validated to the extent that industry can then take the tools and use data specific to their particular design to create a validated model for their specific application.

The RISMC Toolkit quality assurance process includes the activities of verification, validation, assessment, and related documentation to facilitate reviews of these activities. To support activities such as validation, a variety of experimental results will be identified and collected specific to each tool/application. These include (see Figure 3-10) results from facility operation, integral effects test, separate effect tests, and fundamental tests including experiments on individual components. Separate effect test results are used to validate and quantify uncertainty for specific physics models while component test results are used to identify and represent key parameters for component models. For example, tests related to component performance during flooding conditions represent a separate effects test. Integral effects tests are performed on large-scale experimental facilities and can be used to validate how well the code(s) represents typical scenarios that may be found for off-normal conditions.

The INL has facilitates quality software by implementing modern software management processes (including the use of tools such as source code version control), conducting NQA-1 audits, and creating a software verification and validation plan (SVVP). The SVVP identifies software requirements and the associated tests that will be used to validate specific tools. For long-term applications, validation-data support will be a community-scale effort.

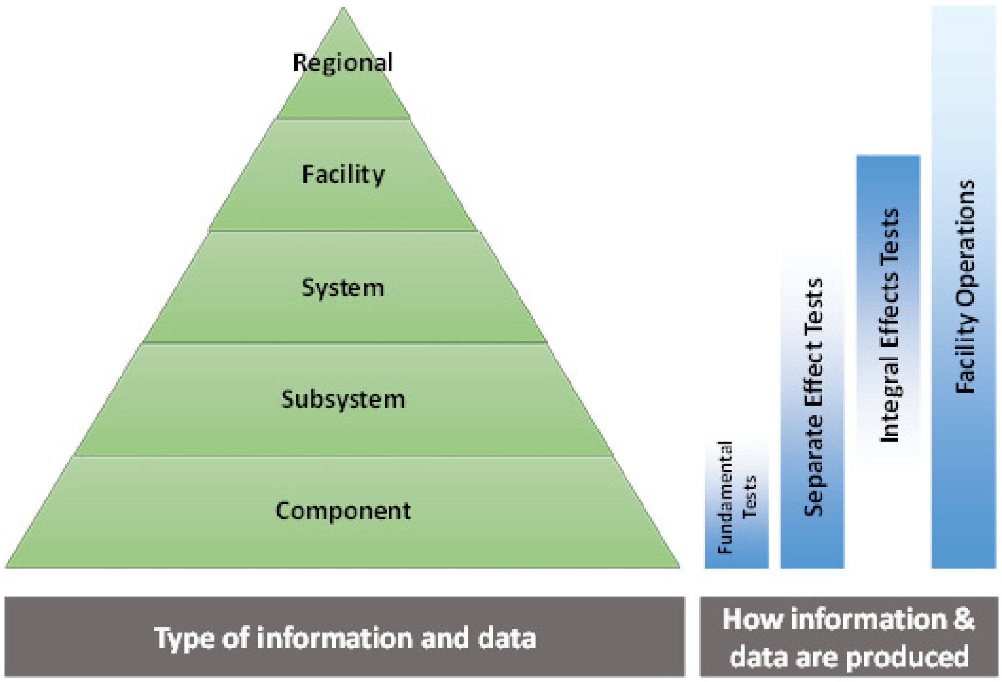

Figure 3-10: Information types and sources that will be used for validation.

For each tool in the RISMC Toolkit, we are in the process of creating a "development plan" (see Table 3-1) that will document a variety of information with the purpose of better understanding and communicating the software development process and outcomes for a specific tool.

Table 3-1. Information captured in the RISMC Toolkit Software Development Plan.

1. Introduction
    a. Overview
        i. Purpose of software and/or tool
        ii. Targeted users
    b. Quality Assurance Requirements/Categorization
2. General Capabilities and Features
3. Development Requirements
    a. Software Design and Structure
    b. Numerical Methods
    c. Physics Implementation
    d. Graphical User Interface
    e. Data Storage and Retrieval
    f. Configuration Management
4. Verification and Validation Plan
5. Development Schedule
    a. Document when specific features are planned to be implemented
    b. Planned releases
6. Technology Transfer
    a. Deployment Plan
    b. Intellectual Property Plan
7. User Documentation

3.2 The RISMC Toolkit

The RISMC Toolkit is being built using the INL's Multi-physics Object Oriented Simulation Environment (MOOSE) HPC framework. MOOSE is the INL development and runtime environment for the solution of multi-physics systems that involve multiple physical models or multiple simultaneous physical phenomena. The systems are generally represented (modeled) as a system of fully coupled nonlinear partial differential equation systems (an example of a multi-physics system is the thermal feedback effect upon neutronics cross-sections where the cross-sections are a function of the heat transfer). Inside MOOSE, the Jacobian-Free Newton Krylov (JFNK) method is implemented as a parallel nonlinear solver that naturally supports effective coupling between physics equation systems (or Kernels). The physics Kernels are designed to contribute to the nonlinear residual, which is then minimized inside of MOOSE. MOOSE provides a comprehensive set of finite element support capabilities (libMesh) and provides for mesh adaptation and parallel execution. The framework heavily leverages software libraries from DOE SC and NNSA, such as the nonlinear solver capabilities in either the Portable, Extensible Toolkit for Scientific Computation (PETSc) project or the Trilinos project.

The RISMC Toolkit provides the foundation for the analysis steps found in the RISMC method. In Figure 3-11, we show the roles that each of the four tools support as part of safety margin analysis.

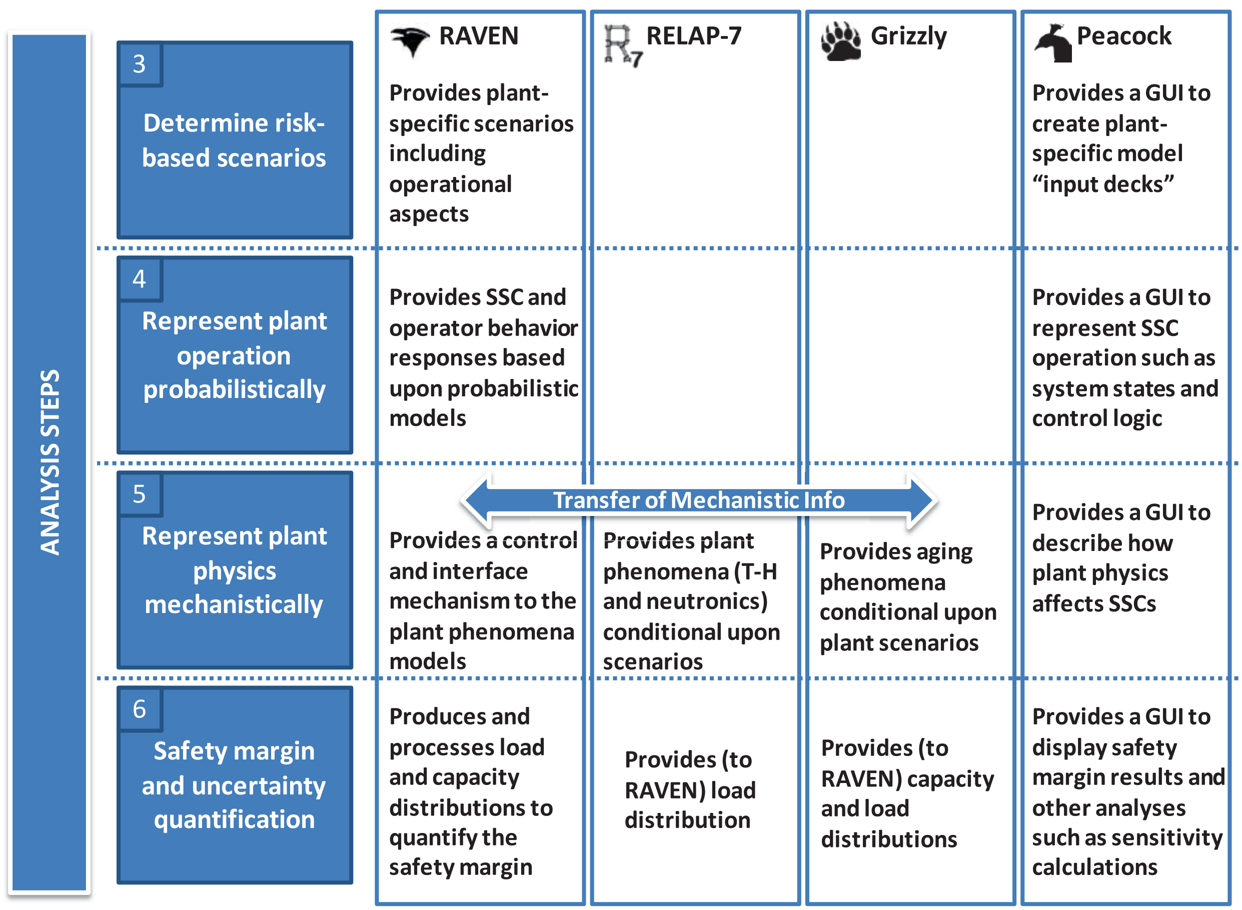

Figure 3-11. The RISMC Toolkit roles in the analysis steps.

3.2.1 RELAP-7

RELAP-7 (Reactor Excursion and Leak Analysis Program-7) is the nuclear reactor system safety analysis code under development within the RISMC Pathway. It is an evolution in the RELAP-series reactor systems safety analysis applications. The RELAP-7 code development is taking advantage of the progresses made in the past three decades to achieve simultaneous advancement of physical models, numerical methods, coupling of software, multi-parallel computation, and software design. RELAP-7 uses the INL's open source MOOSE (Multi-Physics Object-Oriented Simulation Environment) framework for efficiently and effectively solving computational engineering problems. Unlike the traditional system codes, all the physics in RELAP-7 can be solved simultaneously (i.e., fully coupled), resolving important dependencies, and significantly reducing spatial and temporal errors relative to traditional approaches. This allows RELAP-7 development to focus strictly on systems analysis-type physical modeling and gives priority to the retention and extension of RELAP5's system safety analysis capabilities. In addition to the mechanistic calculations that are performed in RELAP-7 to represent plant physics, it has been designed to be integrated into probabilistic evaluation using the RISMC methodology. The RELAP-7 design is based upon:

- Modern Software Design:
  - ✓ Object-oriented C++ construction provided by the MOOSE framework
  - ✓ Designed to significantly reduce the expense and time of RELAP-7 development
  - ✓ Designed to be easily extended and maintained and to couple with other MOOSE modules
  - ✓ Meets NQA-1 requirements
- Advanced Numerical Integration Methods:
  - ✓ Multi-scale time integration, PCICE (operator split), JFNK (implicit nonlinear Newton method), and a point implicit method (long duration transients)
  - ✓ New pipe network algorithm based upon Mortar FEM (Lagrange multipliers)
  - ✓ Ability to couple to multi-dimensional reactor simulators
- State-of-the-Art Physical Models:
  - ✓ All-speed, all-fluid (vapor-liquid, gas, liquid metal) flow
  - ✓ Well-posed 7-equation two-phase flow model
  - ✓ New reactor heat transfer model based upon fuels performance

These features are summarized in Table 3-1.

Table 3-2. Overview of defining RELAP-7 features.

#	CATEGORY	FEATURES	VALUE ADDED
1	**Capabilities**	PWR, BWR, and single- and two-phase fluid system transient thermal-hydraulic analysis available in the same code	• One code with broad capabilities capable of replacing several legacy codes currently in use for design and safety analysis
		Seamless software coupling through MOOSE to other high-fidelity codes: BISON (3D fuel performance); RATTLESNAKE (3D transport neutronics); BIGHORN (3D single- and multi-phase CFD); GRIZZLY (structural analysis)	• Higher fidelity is available (3D models and multi-scale) using other MOOSE based application without need for further software development • Several multi-physics coupling schemes are available and could be fit to the needs dictated by the physics to be modeled
2	**Physics**	Well-posed hierarchy of two-phase flow models; full library of components required to model PWR and BWR systems.	• Permits multi-scale analysis of flow • Eliminates legacy issues with existing codes • Improved modeling of wave effects (waterhammer and BWR instability)
3	**Software Design**	C++ within MOOSE framework. NQA-1 code development process.	• Shorter development cycle • Easier maintenance for code updates
4	**Hardware Platform**	Desktop computers for typical T/H modeling. High Performance Computing for large 3D problems and statistical analyses.	• Graded platform requirements depending on application
5	**Numerical Methods**	Flexible time-integration and spatial discretization methods that enable tight coupling of multi-physics phenomena.	• Second-order accurate in time and space • More robust • Improved run time
6	**Verification**	Strict verification of software, numerical methods and physical models. Regression testing, convergence studies, etc.	• Modern approach to detect and correct code errors promptly • Dramatically reduce code bugs • Uncertainty reduction in software numerical model representation of descriptive physics

RELAP-7 simulates behavior at the plant level with a level of fidelity that will support the analysis and decision-making necessary to economically and safely extend and enhance the operation of the current NPP fleet.

An actual reactor system is complex and contains hundreds of different safety related components. Therefore, it is impractical (and not necessary) to resolve the entire geometry of the whole system in a fully detailed 3D fashion. Instead, representative thermal hydraulic models are used to depict the major physical components and describe major physical processes (such as fluids flow and heat transfer). Consequently, in RELAP-7, there are five main types of components/capabilities:

- Three-dimensional (3D)
- Two dimensional (2D)
- One-dimensional (1D) components – An example of a 1D component is a pipe.
- Zero-dimensional (0D) components for setting boundary conditions for the 1D components – An example of a 0D component would be a pump that provides a pressure (or corresponding fluid flow rate) increase to the pipe connected to the pump.
- 0D components for connecting 1D components

Development progress has been made on RELAP-7. During FY14, a RELAP-7 Theory Manual was completed [10]. Two-phase flow modeling capability has been developed in the code, aimed at demonstrating simulation of a boiling water reactor (BWR) with simplified geometries under extended station blackout (SBO) transient conditions. A number of components developed for single-phase pressurized water reactor model analysis (such as Pipe and Core Channel) have been extended to include two-phase flow modeling capability. Additionally, a set of new components have been developed, including the Separator Dryer, Down Comer, Valve, Turbine, and Wet Well (for example, see the components shown in Figure 3-12). A full seven-equation, two-phase model has been implemented into RELAP-7 and the results have been demonstrated. [11]

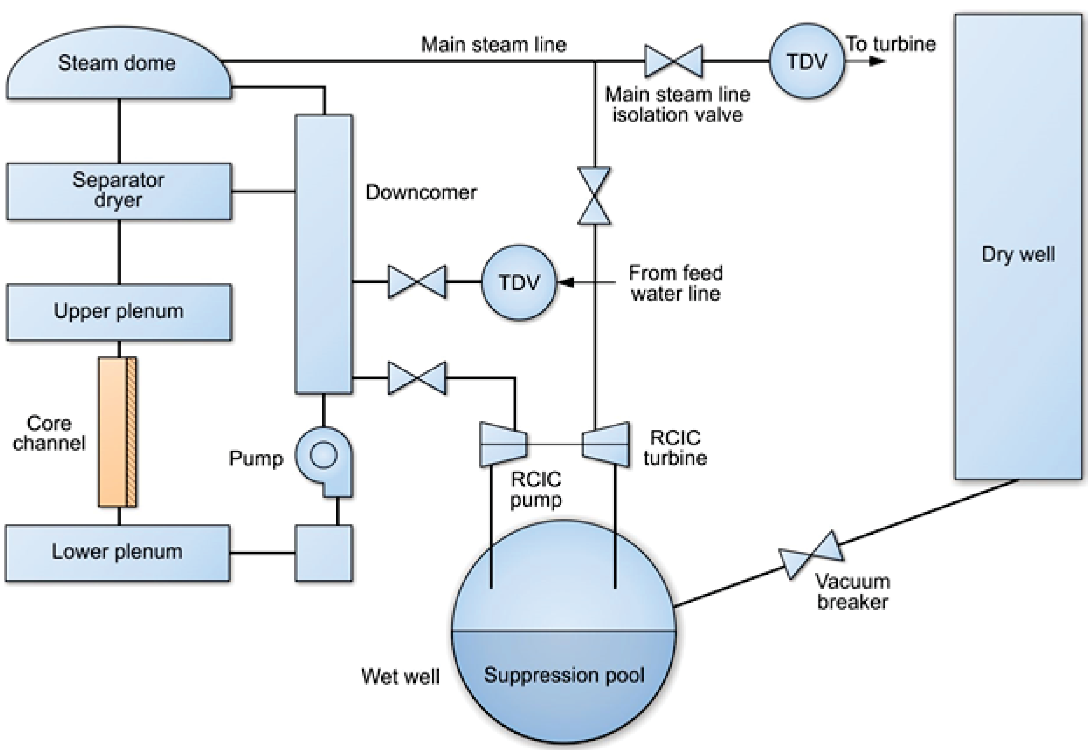

Figure 3-12. Schematic of the components available in RELAP-7 for a BWR model.

Components that are currently in RELAP-7 are shown in Table 3-3 while details of the component features are listed in Table 3-4.

Table 3-3. RELAP-7 components.

Component ID	Description
Pipe	1-D fluid flow within solid structure with wall friction and specified wall temperature or wall heat flux.
PipeWithHeatStructure	1-D fluid flow associated with a 1-D/2-D solid heat structure, including fluid flow, solid heat conduction with different boundary conditions.
CoreChannel	Simulating reactor flow channel and fuel rod, including 1-D flow and 1-D/2-D fuel rod heat conduction.
Subchannel	Simulating 3-D single-phase channel flow (minimal closure relations).
HeatExchanger	Heat exchanger model, including single-phase and two-phase homogeneous equilibrium two-phase flow fluid flow in two sides and heat conduction through the solid wall.
TimeDependent-Volume	Time Dependent Volume, provides pressure and temperature boundaries for other components.
TDM	Time dependent mass flow, which provides mass flow boundary condition.
Branch	Multiple in and out 0-D junction for single-phase and two-phase flow homogeneous equilibrium two-phase flow.
Valve	One in and one out junction with controlled flow area and resistance to simulate valve open / close action for low-speed single phase flow.
CompressibleValve	Simulate valve open and close behavior for compressible flow, including choking for single-phase gas; can be used as safety relief valve.
CheckValve	Simulate the check valve behavior with the form loss calculated by the abrupt area change model
Pump	A junction model with momentum source connecting two components.
PointKinetics	0-D neutron kinetics model.
SeparatorDryer	Separating steam and water with mechanical methods, 1 in and 2 out branches with volume.
Downcomer	0-D volume to mix different streams of water/steam.
WetWell	0-D volume to simulate BWR suppression pool and its gas space.
Reactor	A virtual component that allows users to input time dependent thermal power for CoreChannel model.
Turbine	Simulate BWR Reactor Core Isolation Cooling turbine which drives the pump through a common shaft, 0-D junction.
Accumulator	Simulate a tank to passively provide water to the PWR reactor core during emergencies in which the pressure drops rapidly.

Table 3-4. RELAP-7 component features.

RELAP-7 Component	Dimensionality			Hydrodynamic Model			3D Linkage
	0D	1D	2D	Single Phase	Two Phase HEM	Two Phase 7-Eq.	Application
Pipe	n/a	■	n/a	■	■	■	n/a
PipeWithHeatStructure	n/a	■	■	■	■	■	n/a
CoreChannel	n/a	■	■	■	■	■	BISON
HeatExchanger	n/a	■	□	■	■	□	n/a
TimeDependentVolume	■	n/a	n/a	■	■	□	n/a
TimeDependentMassFlowRate	■	n/a	n/a	■	■	□	n/a
Branch	■	n/a	n/a	■	■	□	n/a
Valve	■	n/a	n/a	■	□	□	n/a
CompressibleValve	■	n/a	n/a	■	□	□	n/a
CheckValve	■	n/a	n/a	■	□	□	n/a
Pump	■	n/a	n/a	■	□	□	n/a
PointKinetics	■	n/a	n/a	n/a	n/a	n/a	n/a
SeparatorDryer	■	n/a	n/a	n/a	■	□	n/a
Downcomer	■	n/a	□	n/a	■	□	n/a
WetWell	■	□	n/a	■	■	□	n/a
Reactor	■	n/a	n/a	n/a	n/a	n/a	Rattlesnake+ MAMMOTH
Turbine	■	n/a	n/a	■	□	□	n/a
Pressurizer	■	□	n/a	n/a	■	□	n/a
Accumulator	■	□	n/a	■	■	□	n/a

In summary, RELAP-7 provides a unique and advanced system-safety analysis capability that:

- Represents a variety of components for pressurized and boiling water reactors
- Uses single- and two-phase flow models (user selectable)

- Leveraged advances in computational sciences for advanced numerical methods including parallel processing on high-performance computers

- Facilitates a multi-physics capability by coupling with other mechanistic codes within the MOOSE framework

3.2.2 RAVEN

The Risk Analysis in a Virtual ENvironment (RAVEN) is the module tailored to RELAP-7 that controls the risk simulation, including the generation of accident scenarios, and the thermal-hydraulics that evolves during a simulation. Historically, in older thermal-hydraulics codes, the plant control logic model controller is separate from the thermal-hydraulic solver used by the systems-analysis code. The reason for this choice is that most of the time the mathematical representation of the control logic involves discrete function that are not suitable for the numerical thermal-hydraulic solver.

However, by tightly coupling RAVEN with RELAP-7, INL has created a unique dynamic modeling capability that provides a higher fidelity in scenario representation and control. For example, as part of the RELAP-7 analysis, it is now possible to introduce complex behaviors such as component failures, subsequent component recoveries, and plant feedback based upon plant conditions. Further, dynamic behavior can be represented such as component failures that may depend on time-depended plant "signals" (or observable physics such as pressure and temperature). As a consequence of this new capability, the scenario generator is more a scenario controller where we can integrate probabilistic behavior with the mechanistic analysis provided by RELAP-7. [12]

What enables this tight coupling of RAVEN and RELAP-7 is the MOOSE framework. RAVEN can control, through the system of equations managed by MOOSE, the plant thermal hydraulics.

This RAVEN and RELAP-7 flexibility could be used also to import proprietary correlations or control laws without the need to develop an ad-hoc derivative version of the code – these extensions can simply be included as needed in proprietary calculations. Further, the control logic is understandable as compared to traditional approaches, where "compactness" of the input deck seemed to be the driving factor, resulting in very complicated (to create and understand) input structures.

Currently, RAVEN has the capability to "drive" MOOSE-based applications such as RELAP-7 for which the following functional capabilities are provided:

- Front end driver for RELAP-7:
    - Plant description to RELAP-7 (component, control variable, and control parameters)
    - Runtime environment
    - Parallel distribution of RELAP-7 runs (adaptive sampling)
- Control logic required to:
    - Simulate the reactor plant control system
    - Simulate the reactor operator (procedure guided) actions
    - Perform Monte Carlo sampling of random distributed events
    - Perform event tree based analysis
- GUI (underlying infrastructure is provided by Peacock) to:
    - Concurrent monitoring of control parameters
    - Concurrent alteration of control parameters
- Advanced analysis capabilities
    - Adaptive sampling
    - Dynamic scenario generation and evaluation

    - Limit surface determination
    - Emulators or reduced-order models
- Post Processing data mining capability based on:
    - Dimensionality reduction
    - Cardinality reduction
    - Uncertainty quantification and propagation

During FY15, a RAVEN User's Manual was produced and is available for users. [13] This manual provides information related to RAVEN including install the software, running the software, technical details of features, and examples.

3.2.3 Grizzly

Grizzly is a MOOSE-based tool for simulating component ageing and damage evolution events for LWRS specific applications. Grizzly will have implicit time simulation capabilities for component damage evolution concerning LWR pressure vessel, core internals, and concrete support and containment structures subjected to a neutron flux, corrosion, and high temperatures and pressures. Grizzly will heavily leverage the thermo-mechanics physics found in the BISON fuels performance application as a starting point. [14] Grizzly will be able couple with RELAP-7 and RAVEN to provide aging analysis in support of the RISMC methodology.

An initial proof of concept demonstration has been performed to demonstrate Grizzly's ability to model a degraded reactor pressure vessel under pressurized thermal shock (PTS) loading conditions. A full 3D model of a PWR reactor pressure vessel (without the stainless steel liner) was subjected to uniform thermal and pressure boundary conditions on the inner surface under two postulated accident scenarios. Grizzly was used to solve the coupled thermal and mechanical response of the system. In addition, the model of Eason, Odette, Nanstad and Yamamoto, [15] referred to here as the EONY model, has been incorporated into Grizzly to predict the embrittlement of the RPV steel due to thermal aging and radiation damage. This model represents the degradation as a shift in the ductile to brittle transition temperature of the steel, and is based on experimental data, which only goes back to the lifespan of the longest-operating reactors.

Current work on Grizzly has focused on two goals:

1. To enhance Grizzly to permit it to be used for an engineering scale fracture assessment of an embrittled RPV.

2. To develop methods to characterize material evolution during irradiation and thermal aging in order to characterize embrittlement for time frames beyond the lifetime of the current reactor fleet using science-based methods (rather than relying solely on data).

In FY14, the initial investigation into concrete aging started [16]. Concrete is a primary material used in the construction of NPPs because of its structural strength and its radiation shielding ability. As such, when life extension is considered, it is critical to have predictive tools and methods specific to aging of concrete structures. Long-term degradation of concrete structures in nuclear power plants is mainly influenced by different (and possibly interacting) processes, including physical (e.g., elevated temperature, radiation), chemical (e.g., slow hydration, leaching) and mechanical (e.g., cracking).

3.2.4 Peacock (demonstrated using RAVEN)

The Peacock software is a general Graphical User Interface (GUI) for MOOSE based applications. Peacock has been built in a general fashion so to allow specialization of the GUI for different applications via an Application Programming Interface (API). The specialization of Peacock for RELAP-7/RAVEN allows both a graphical input of the RELAP 7 input file, online data visualization and is moving forward to provide a direct user control of the simulation and data mining capabilities in support of PRA analysis.

The Peacock realization for RAVEN has four main tabs: Input File, Execute, Postprocess, and Visualize. A short description of each tab is below reported:

- Input File – This option (Figure 3-13) provides the interface for creating the input for a typical plant. On the far left a tree menu allows creating the input for each component of the plant as also for the general simulation setting. On the right the whole plant is pictured, components are shown as soon as added to the input. Peacock has the capability to access to the input description of a component already created by selecting (via a double click) its visual representation.

- Execute – This is the windows running and monitoring the simulation. In RAVEN, it shows buttons that open the capability to set up the parameters for parallel sampling and parallel running of the simulation. The large central box is used to collect the input coming out directly from the simulation. For a single run this input is directly the RELAP-7 output while for multiple runs is the output from the RAVEN simulation control.

- Postprocess – This option is used to visualize every variable exported by the simulation in CVS format (comma separated value). The variable that could be exported in this format are: monitored, controlled, and auxiliaries.

- Visualize – This option (Figure 3-14) allows visualization of the RELAP-7 solution while the simulation is running. The solution (e.g. temperature, pressure, velocity) are projected on a plant diagram and the time changes are visualized in a movie-like fashion. The visualization allows users to move to any point in time already simulated to re-examine the time evolution.

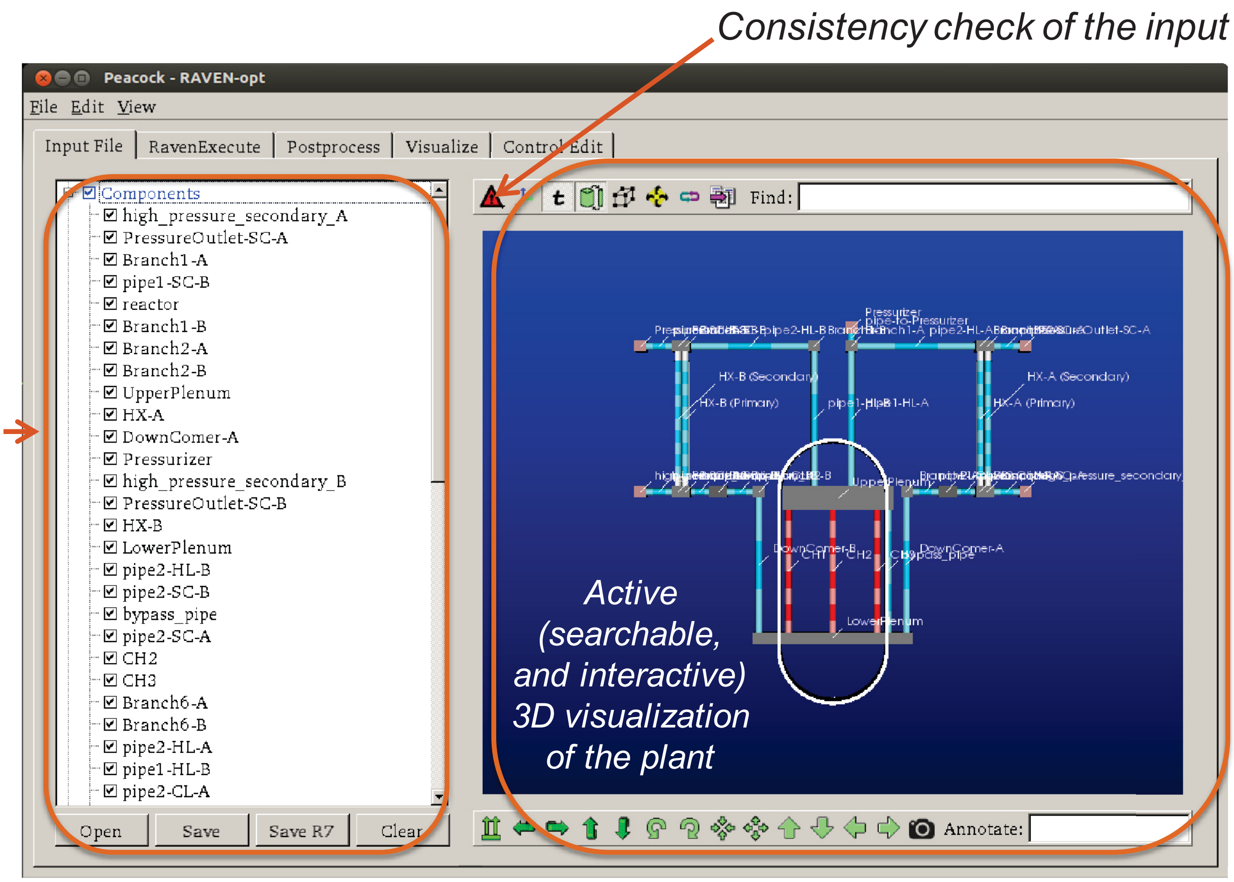

Figure 3-13. Representation of the Peacock graphical user interface for input.

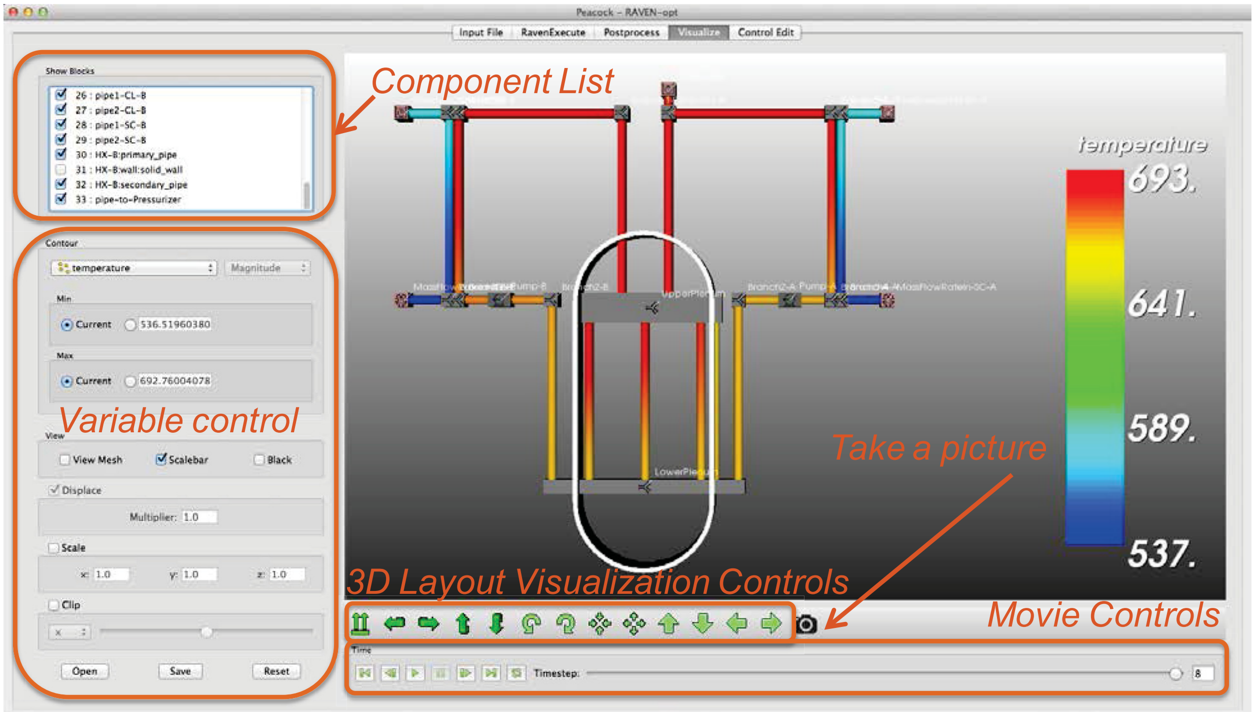

Figure 3-14. Representation of the Peacock results visualization.

3.3 R&D Collaborations

A variety of avenues are being followed in order to foster collaborations within the RISMC Pathways, including:

- RISMC Methodology and Toolkit is being developed together with EPRI
- Research results are disseminated via a variety of technical meetings, conferences, and are made available in program reports
- Industry is the targeted users group for RISMC Tools
- Code “testers” are being actively sought, the RISMC tools will be made available to industry volunteers who will use the tools and provide feedback to the LWRS Program
- A “User’s Group” is being considered for maintenance and application information

3.3.1 EPRI Collaboration

EPRI has established the Long-Term Operations Program, which complements the DOE LWRS Program. EPRI’s and industry’s interests include applications of the scientific understanding and the tools to achieve safe, economical, long-term operation of the operational fleet of NPPs currently in service. Therefore, the government and private sector interests are similar and interdependent, leading to strong mutual support for technical collaboration and cost sharing. The interface between DOE-NE and EPRI for R&D work supporting long-term operation of the existing fleet is defined in a memorandum of understanding between the two parties. A joint R&D plan defining the collaborative and cooperative

R&D activities between the LWRS Program and the Long-Term Operations Program has been developed. Also, contracts with EPRI or other industrial organizations may be used as appropriate for some work.

3.3.2 University and Regulatory Engagement

Universities participate in the LWRS Program in at least two ways: (1) through the Nuclear Energy University Program (NEUP) and (2) via direct contracts with the national laboratories that support the Program's R&D objectives. NEUP funds nuclear energy research and equipment upgrades at U.S. colleges and universities and provides scholarships and fellowships to students (see www.neup.gov). In addition to contributing funds to NEUP, the LWRS Program provides descriptions of research activities important to the LWRS Program and the universities submit proposals that are technically reviewed. The top proposals are selected and those universities then work closely with the LWRS Program in support of key LWRS Program activities. Universities also are engaged in the LWRS Program via direct subcontracts where unique capabilities and/or facilities are funded by the program.

Recent and current NEUP-funded projects that interact with the RISMC Pathway are:

- 11-3030 with Professor Tunc Aldemir at the Ohio State University. The focus of this research was on passive component reliability modeling in a multi-physics simulation environment and was started in 2011.

- 13-5142 with Professor Halil Sezen at the Ohio State University. The focus of this research is on the creation of an approach to external events PRA for structures and components and the associated integration into an existing risk assessment. Case studies are being evaluated in order to implement new external events approaches (for example, for seismic events) into the MOOSE platform.

- 14-6442 with Professor Klein at Oregon State University. The focus of this research is on probabilistic economic valuation of safety margin management. The goal of the project is to use the RISMC Toolkit (e.g., RAVEN, RELAP-7) to perform probabilistic assessment of accident scenario consequences and costs avoided by safety margin upgrades and to compare to costs of safety margin upgrade installation for cost/benefit analysis. This work will provide risk-informed, data-driven decision making, for both plant owners and regulatory bodies

DOE's mission to develop the scientific basis to support both planned lifetime extension up to 60 years and lifetime extension beyond 60 years, and to facilitate high-performance economic operations over the extended operating period for the existing LWR operating fleet in the United States, is the central focus of the LWRS Program. Therefore, more and better coordination with industry and NRC is needed to ensure a uniform approach, shared objectives, and efficient integration of collaborative work for the LWRS Program. This coordination requires that articulated criteria for the work appropriate to each group be defined in memoranda of understanding that are executed among these groups. NRC has a memorandum of understanding in place with DOE, which specifically allows for collaboration on research in these areas. Although the goals of the NRC and DOE research programs may differ, fundamental data and technical information obtained through joint research activities are recognized as potentially of interest and useful to each agency under appropriate circumstances. Accordingly, to conserve resources and to avoid duplication of effort, it is in the best interest of both parties to cooperate and share data and technical information and, in some cases, the costs related to such research, whenever such cooperation and cost sharing may be done in a mutually beneficial fashion.

3.4 RISMC Advisory Committee

The RISMC Pathway Advisory Committee consists of a collection of individuals with backgrounds (knowledge and skills) from academia, nuclear consultants, owner/operators, and vendors that can assist in providing:

- Technical review and guidance on the scientific methods and tools:
    - Being developed now and considered in the future as part of the RISMC Pathway
    - Being validated for use in industry applications
- Strategic guidance on the overall focus areas to be considered in the RISMC Pathway, including:
    - Recommendations on high-value applications of interest the RISMC stakeholders
    - Review and feedback on the RISMC Pathway Plan
    - Recommendations on communication of the RISMC value and technical achievements

In FY14, the initial structure and member selection began. The Committee was formally started in FY15 and held its initial kick-off meeting in July, 2015. Feedback from this Committee will be used to further modify and refine the Pathway Plan. Members (as of FY15) of the Committee are:

- George Apostolakis, Consultant
- Doug True, Consultant
- John Gaertner, Consultant
- Enrico Zio, Professor Politecnico di Milano, Italy
- Kord Smith, Professor Massachusetts Institute of Technology
- Greg Krueger, Exelon Director of Risk Management
- Gabe Balog, Candu Owners Group Director of R&D
- Bret Boman, Areva

4. RESEARCH AND DEVELOPMENT COOPERATION

4.1 DOE Collaborations

4.1.1 Nuclear Energy Advanced Modeling and Simulation (NEAMS).

The NEAMS Program is developing a simulation tool kit that will accelerate the development and deployment of nuclear power technologies that employ enhanced safety and security features, produce power more cost-effectively, and utilize natural resources more efficiently. The overall objective of NEAMS is to develop and validate predictive analytic computer methods for the analysis and design of advanced reactor and fuel cycle systems. The LWRS Program intends to take advantage of the detailed, multiscale, science-based modeling and simulation results developed by the NEAMS Program. The modeling and simulation advances will be based on scientific methods, high dimensionality, and high-resolution integrated systems. The simulations will use the most advanced computing programs and will have access to the most advanced computation platforms that are available to DOE. These tools will include fully three-dimensional, high resolution, representation of integrated systems based on physical models. Included in these tools will be safety codes integrated predictive physics for nuclear fuels, reactor systems, and separations processes.

4.1.2 Consortium on Advanced Simulation of LWRs (CASL).

The CASL Hub is the first DOE Energy Innovation Hub established in July 2010, for the purpose of providing advanced modeling and simulation (M&S) solutions for commercial nuclear reactors. The main focus is on ultra-high fidelity of reactor core physics. CASL is developing a detailed model of the LWR core; if investigations in the LWRS Program warrant it, the LWRS Program-developed models can couple with the CASL-developed models. CASL has an interest in using RELAP-7 for one or more of their challenge problems. The RISMC Pathway will be collaborating with CASL on aspects of technology transfer to industry related to methods and tools being developed in the respective programs.

4.1.3 Nuclear Energy Enabling Technologies (NEET)

The NEET Program is developing crosscutting technologies that directly support and complement the DOE advanced reactor and fuel cycle concepts, focusing on innovative research that offers the promise of dramatically improved performance. It coordinates research efforts on common issues and challenges that confront the LWRS, Advanced Reactor Technologies (ART), and Small Modular Reactors (SMR) to advance technology development and deployment.

4.2 Industry Interactions

Industry is significantly engaged in RISMC activities, and the level of engagement is increasing. Up to now, industry engagement in RISMC (primarily through EPRI) has taken place at two levels: (1) input into program planning and (2) active participation in RISMC activities. One effect of this influence has been strengthening the RISMC team consensus that RISMC developments should be driven by “use cases” (i.e., explicitly planned eventual applications that are used to formulate requirements on development of the next-generation capability) and “case studies” (i.e., actual applications that scope particular developments and, once completed, support assessment of the current phase of development). EPRI and other industry representatives are becoming increasingly involved in detailed technical planning of Industry Applications that now drive development activities and are expected to continue to support actual execution. This has two effects: (1) it helps to ensure the program moves in a direction that addresses practical industry concerns, and (2) it provides the RISMC team with access to engineering expertise that is needed in development of enabling methods and tools.

Coordination of RISMC activities includes the following:

- **EPRI:** EPRI will continue to play an important role in high-level technical steering and in detailed planning and execution of RISMC case studies. EPRI also will play a critical role in engaging industry stakeholders (i.e. personnel form operational NPPs) to support pathway development, contribute technical expertise to use case development and evaluate technical results from case study applications. The RISMC Pathway R&D is coordinated with EPRI Long-Term Operation Program work.

- **Owners Groups:** Interactions will continue with groups such as the CANDU, BWR, and PWR Owners Groups through information exchange and evaluations of specific topics via case studies. Recent technical exchange meetings have included participants from both Westinghouse and GE Nuclear. In addition, staff from the CANDU Owners Group have been trained on the MOOSE Framework and will be interacting with INL analysts and developers that are working on the RISMC Pathway.

- **Other industry partners:** Involvement of engineering and analysis support from industry is presently foreseen in the performance of case studies to drive next-generation analysis development and in formulation of component models for implementation in next-generation analysis capability. The individuals prospectively involved are either industry consulting firms or currently independent consultants who have working relationships with current licensees. All individuals are experts in applying traditional safety analysis tools and are conversant with risk-informed analysis.

- **Multilateral International Collaboration**: A variety of international researcher interactions are of potential interest to the RISMC Pathway, including:

    - The Committee on the Safety of Nuclear Installations (CSNI). This committee is a Organization for Economic Co-operation and Development (OECD)-sponsored group that is part of the Nuclear Energy Agency. One of the task groups in CSNI was focused on Safety Margin Applications and Assessment. The Working Group on Risk Assessment (WGRISK) advances the understanding and use of PRA tools. The Working Group on Analysis and Management of Accidents addresses safety analysis research including the uncertainty and sensitivity evaluation of best-estimate methods program. Various benchmarking activities are organized. Meetings are held twice a year in Paris in June and December. DOE (Rich Reister) is a member. A second Working Group is the newly-formed Working Group on Natural External Hazards (WGEV). The mission of the WGEV is to improve the understanding and treatment of external hazards that would support the continued safety performance of nuclear installations, and improve the effectiveness of regulatory practices. INL (Curtis Smith) is a member of this Working Group.

    - The European Nuclear Plant Life Prediction (NULIFE) – A virtual organization funded by over 50 organizations and the European Union under the Euratom Framework Program. This organization is working on advancing safety and economics of existing NPPs.

- The Advanced Safety Assessment Methodologies: Extended PSA (ASAMPSA_E) organization is investigating challenging initiating events such as the combination of two correlated extreme external events (earthquake and tsunami). The consequences of these situations, in particular flooding, has the potential to go beyond what has been considered in some NPP designs. Such situations can be identified using probabilistic safety assessment (PSA) methodology that complements the deterministic approach for beyond design accidents. The ASAMPSA_E group aims at promoting good practices for the identification of such situations with the help of PSAs and for the definition of appropriate criteria for decision-making in the European context. It offers a new framework to discuss, at a technical level, how "extended PSA" can be developed efficiently and be used to verify if the robustness of Nuclear Power Plants (NPPs) in their environment is sufficient. The project has experts from 28 organizations in 18 European countries. Members of the RISMC project team have interacted with this organization, including attending the first End Users workshop in May 2014.

- Civil Nuclear Energy Research and Development Working Group (CNWG), which is a bilateral activity with Japan that is focusing on RISMC and advanced seismic PRA applications. Tasks included in the CNWG include participation in a PRA expert's roundtable wherein U.S. participants discussed how PRA is applied in the U.S. and the resulting benefits, and Japanese participants discuss the issues associated with PRA application in Japan. Also advanced seismic PRA and RISMC activities are being considered for additional activities.

4.3 Industry Applications

One of the primary avenues for collaboration with industry is through the RISMC Industry Applications. The primary purpose of Industry Applications in the RISMC Pathway is to demonstrate advanced risk-informed decision making capabilities for relevant industry questions. The end goal of these activities is the full adoption of the RISMC tools and methods by industry applied to their decision making process.

The elements of the above proposition are further explored below:

(a) Demonstrate
- Provide confidence and a technical maturity in the RISMC methodology (essential for broad industry adoption)
- Strong stakeholder interaction required
- Address a wide range of current relevant issues (see also item (d))

(b) Advanced
- Analyze multi-physics, multi-scale, complex systems
- Use of a modern computational framework
- A variety of Methods, Tools, and Data can be utilized (e.g. use of legacy tools and state-of-the-art tools as they become available for use)
- Be as realistic as practicable (with the use of appropriate supporting data)
- Consider uncertainties appropriately and reduce unnecessary conservatism when warranted

(c) Risk-Informed decision making capabilities
- Use of an integrated decision process
- Integrated consideration of both risks and deterministic elements of safety

(d) Relevant industry applications

- Currently, the RISMC Pathway has identified four Industry Applications (IA) to cover a range of current industry issues (in order of importance):
  - IA1 – Performance-Based ECCS Cladding Acceptance Criteria
  - IA2 – Enhanced External Hazard Analyses (multi-hazard)
  - IA3 – Reactor Containment Analysis
  - IA4 – Long Term Coping Studies/FLEX

Each Industry Application covers a broad range of relevant plant technical issues. These issues are discussed in a prior report INL-EXT-14-32928 [17], where we selected, prioritized, and combined important plant issues into the four above Industry Application categories. These are the most relevant industry topics of today that can potentially impact plant operations in a significant way, in the near future, making them interesting, relevant, applications for the RISMC toolkit. Because of their broad range of applicability, each Industry Application may spawn one or more demonstration problems, each depending on stakeholder interest on different aspects of a given IA.

4.3.1 IA1 Performance-Based ECCS Cladding Acceptance Criteria

The RISMC toolkit is now sufficiently matured to offer a potential solution to the loss-of-coolant accident (LOCA) problem and provide to the plant operator a vehicle to manage the margins and inform decisions when compliance with 10 CFR 50.46 is challenged by changes in the operational envelope.

This compliance issue is the driver behind the RISMC Industry Application 1 (IA1), where margin is here relative to the 10 CFR 50.46 rule which is expected to be amended in 2016. The industry will need to comply with the new rule within the following four to six years (the timeline for implementation is still being discussed among the industry stakeholders, and will depend on many factors, such as methodology changes, amount of work to be submitted for regulatory approval, and regulatory reviews).

A LOCA safety analysis involves several disciplines which are computationally (externally) coupled to facilitate the process and maintenance of legacy codes and methods. The key disciplines involved in a LOCA analysis are:

- Core physics (fuel and core design)
- Fuel rod thermos-mechanics
- Clad corrosion
- LOCA thermal-hydraulics

The proposed rule (10 CFR 50.46c) would replace the prescriptive analytical requirements of the previous rule [peak clad temperature (PCT)<2200°F, MLO<17%, etc.], with performance-based requirements. The US NRC Draft Regulatory Guide DG-1263 defines an acceptable analytical limit on peak cladding temperature and integral time at temperature for the zirconium-alloy cladding materials tested in the NRC's LOCA research program. This analytical limit is based on the data obtained in the NRC's LOCA research program.

Referring to DG-1263, the analytical limit presented in Figure 4-1 will substitute for the 17% embrittlement limit. The hydrogen content depends on the burnup value and material characteristic of the cladding, i.e. performance to embrittlement under irradiation for a specific cladding alloy. Note that the temperature limit (at least when the pre-transient hydrogen content is less than 400 weight-ppm) is still the same, i.e. 2200°F. However the margin to embrittlement significantly decreases as the fuel is irradiated in the core and the cladding hydrogen concentration increases.

As a result of local oxidation, a measure of time-at-temperature is anticipated to be the controlling figure-of-merit under the proposed rule. In general terms, the two criteria embrittlement oxidation limit and PCT should be treated jointly.

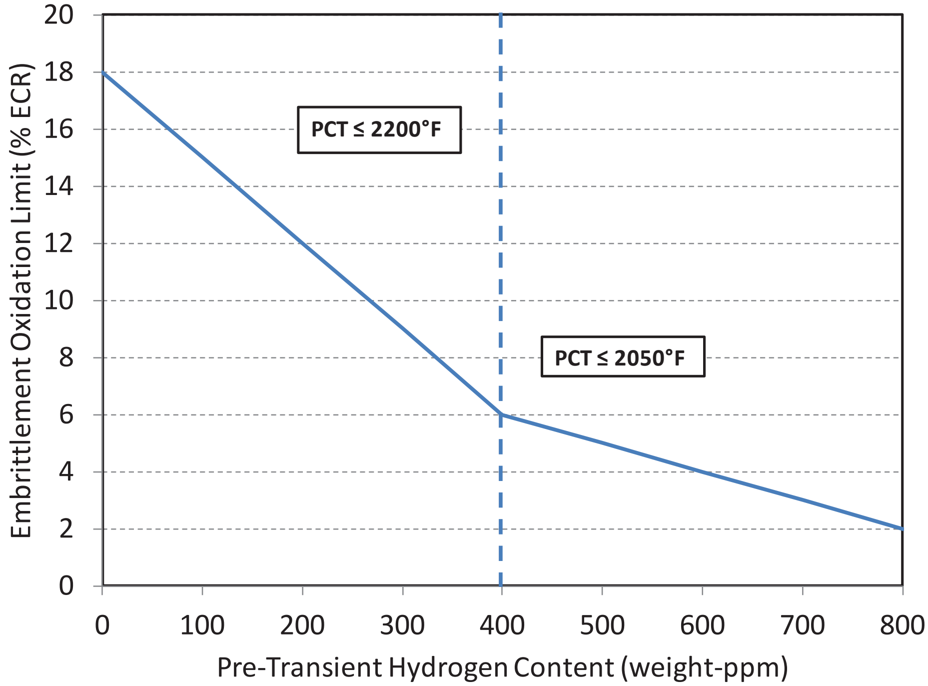

Figure 4-1. Proposed limit on peak cladding temperature and embrittlement oxidation limit [18].

4.3.1.1 IA1 Problem Statement

The RIMM IA1 methodology and tool will provide a means of quantifying the impact on the key LOCA analysis figure-or-merits PCT, equivalent cladding reacted (ECR), and core-wide oxidation (CWO) of a change in LOCA analysis inputs. This information would be obtained without the resource requirement, cost, and schedule, of an actual LOCA reanalysis using a LOCA evaluation model. The information that the tool provides can then be used for decision making and margin management.

The vision for the RIMM IA1 methodology and tool is summarized in the following propositions:

- Provide a responsive toolbox for the plant operator which enables rapid decisions on considered changes within the LOCA issue space (as regulated under the proposed 10 CFR 50.46c). The goal is to greatly reduce the response cycle.
- Enable current knowledge to be factored into the process to enhance safety and operation optimization.
- Quantify currently-unquantified uncertainties to the extent practical.
- An approach that can lead to new knowledge and understanding of the LOCA scenario which could be "locked" in the engineering assumption of licensing calculations. Enable a more effective and optimal engineering exploration of the issue space.

- A "plug-and-play" design of the multi-physics tool which enables plant owners and vendors to consider and further develop the RIMM IA1 Framework for use with their established codes and methods.

4.3.1.2 IA1 Characterization Problem

As a first step, the owner/operator will use the IA1 tool to "characterize" the core designed for operation. Figure 4-2 illustrates this process, where the IA1 tool maps an envelope of maximum embrittlement oxidation limit as a function of cycle exposure. This allows the operator to have a realistic assessment of an operating core, and conceivably be more prepared for a quick response re-analysis in case a problem might occur.

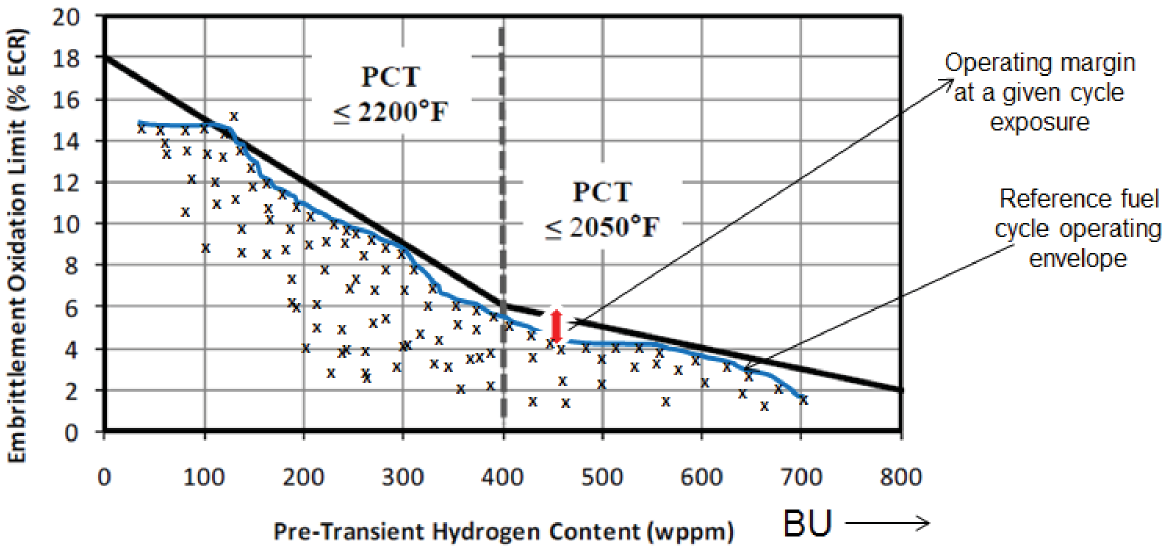

Figure 4-2. Characterization of a hypothetical core (points and curves displayed are notional values, i.e., they are not actual calculations, they are representations of a illustrative outcomes). [19]

4.3.2 IA2 Enhanced Seismic / External Hazard Analysis

Given that hazards external to a nuclear facility may negatively impact a variety of SSCs from direct damage (e.g., failure during a fire) or indirect damage (e.g., consequential failure from a flood following a pipe break), there is a possibility that initiating events, reduced redundancy levels, reduced reliability, or degraded safety barrier may be realized, thereby increasing the likelihood or severity of potential accident scenarios.

A class of hazards to nuclear facilities originates external to the plant. These external events are a class of initiating event that has the initial deviation caused by a hazard located outside the normal plant SSCs. Physical impacts such as fires, floods, and earthquakes are typically included in this group of initiating events. Additional detail on the modeling of external hazards was provided in Section 3.1.3.4.

4.3.3 IA-3 Reactor Containment Analysis

One of the safety improvements mandated by the NRC following the accident at the Fukushima Daiichi nuclear facility is to have reliable, hardened containment venting systems capable of operating under beyond-design-basis (BDA) and severe accident (SA) conditions and installation of containment engineered vent filtration systems to reduce the release of radioactive materials should a SA occur for Mark I and Mark II containments. [20] Given the relatively small volumes of Mark I and II containments which depend on suppression pools and have no mitigation for hydrogen, ensuring the availability of reliable, hardened containment vents may provide plant operators with improved methods to vent containments during wide range of BDA accidents (but before core melt). However, the industry has stated that the addition of filters to hardened containment vents may require modifications to vent design. An EPRI study indicated that the containment venting alone is not effective. It has to be combined with active debris cooling to be effective. [21] Hence, accident sequences need to be better understood to determine under what conditions the filters are beneficial or non-beneficial.

A containment analysis module, coupled to the RELAP-7 code, will be developed to have multi-physics and multi-scale simulation capability with the goal to greatly reduce uncertainty in containment safety analysis and have science-based predictability in safety transient behaviors. In this module, a multi-dimensional analysis capability will be developed to analyze large open spaces within a containment or confinement building to replace traditional lumped parameters approach or pseudo two-dimensional field simulation. Three-dimensional hydrogen transport and detonation capability and fission product transport and deposition capability will be developed with emphasis on verification and validation.

4.3.4 IA-4 Long Term Coping Studies

The "Diverse and Flexible Coping Strategies" (FLEX) [22] aim at increasing defense-in-depth for beyond-design-basis scenarios to address an extended loss of off-site power and loss of normal access to the ultimate heat sink occurring simultaneously at all units on a site. The objective of FLEX is to establish an indefinite coping capability to prevent damage to the fuel in the reactor and to maintain the containment function by using installed equipment, on-site portable equipment, and pre-staged off-site resources. The coping can be thought of as occurring in three phases:

Phase 1: Cope relying on installed plant equipment
Phase 2: Transition from installed plant equipment to on-site FLEX equipment
Phase 3: Obtain additional capability and redundancy from off-site equipment until power, water, and coolant injection systems are restored or commissioned.

The primary objective of establishing the FLEX analysis capability is to establish a RISMC framework which uses the system safety analysis tools to:

1. Better understand the accident sequence and recovery strategies

2. Search any vulnerability that might exist with FLEX.

The FLEX case study requires coordination with the external hazard analysis and containment analysis case studies. The external hazard analysis and containment analysis case studies emphasize more on the deterministic analysis tools development while the FLEX case study emphasizes on RISMC probabilistic methodology development and applications.

5. RESEARCH AND DEVELOPMENT PRODUCTS AND SCHEDULES

The RISMC Pathway will deliver the following high-level products:

1. Technical-basis reports for RIMM
2. The RISMC Toolkit

It has been determined that the focus in the near term will be on NPP Industry Applications that study a specify scope of phenomena, components, and simulation capabilities needed to address the given issue space. As part of these applications, refinement of the associated methods and tools would continue at a reduced level of effort compared to the effort associated with RISMC Toolkit development.

As the development and capabilities of the RISMC Toolkit progresses, INL will collaborate with industry to determine how to transition tools such as RAVEN, Grizzly, and RELAP-7 to a user-supported community of practice, including planning for lifecycle software management issues such as training, software quality assurance, and development support. For example, application of test and operating data to RELAP-7 evaluation was started in FY2015 with data used to validate existing safety analysis codes. As newer data become available to address issues not covered by the old data, comparison with those data will support RELAP-7 refinement.

Assuming a funding profile commensurate with that in the current program plan, RELAP-7 development is expected to be substantially complete as of the December 2014 initial beta release. This does not mean that RELAP-7 would be frozen as of FY2015, any more than previous-generation safety analysis codes have been frozen, but its development would be more evolutionary in nature. The primary objective of the December 2014 beta release is to get feedback and suggestions for improvement on usability and applicability from the user community. Therefore, this beta release will be limited to select users who are experienced in developing and using reactor systems safety analysis codes such as RELAP5 and TRAC.

5.1 Integrated Project Plan Milestones

The major project plan milestones are listed by FY below:

FY2016

- Release the beta version 1.0 of Grizzly. This will include engineering fracture analysis capability for RPVs, with an engineering model for embrittlement, and a modular architecture to enable modeling of aging mechanisms.

- Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for performance-based ECCS cladding acceptance criteria

- Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for enhanced external hazard analyses (seismic and flooding). Release the beta version of initial flooding model.

- Develop an initial margin analysis tool to evaluate reactor containment performance to evaluate the reliability of proposed industry BWR hardened venting systems

- Complete an investigation into the use of the RISMC methodology to validate traditional PRA models.

- Extend RAVEN to provide an emulator capability for complex systems and an optimization search support for risk-informed margin recovery.

- Complete the optimized and validated version of RELAP-7 that provides a tool coupled to RAVEN and to other applications (e.g., aging and fuels modules), used to perform as a balance-of-plant capability for the multidimensional core simulators.

- Beta 1.5 release of RELAP-7 with improved closure relationships and steam/water properties, completed LWR 0D components (such as jet pump and accumulator), improved LWR components (1D-2D downcomer, 1D pressurizer, optional steam generator designs such as helical), tightly coupled multi-physics fuels performance (NEAMS code BISON), and single-phase 3D subchannel flow capability.

- Validation and benchmarking of Grizzly will be conducted for reactor metal applications.

- Validation and benchmarking of Grizzly for concrete aging will be started.

FY2017-2018

- Complete the technical basis reports for Risk-Informed Margins Management.

- Complete a full-scope margins analysis of a commercial reactor power uprate scenario using RELAP-7/RAVEN. Use margins analysis techniques, including use of RELAP-7/RAVEN/Grizzly (component aging module)/others, to analyze an industry-important issue (e.g., assessment of major component degradation in the context of life extension or assessment of the safety benefit of advanced fuel forms). Test cases will be chosen in consultation with external stakeholders.

- Release the beta version 2.0 of Grizzly. This version will include capabilities for modeling selected aging mechanisms in reinforced concrete and for engineering probabilistic RPV fracture analysis.

- Validation and benchmarking of Grizzly for concrete aging will be completed.

- Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for reactor containment analysis including hardened reliable vents and shallow- and deep-water flooding and seismic events.

- Start the demonstration of the margins analysis techniques, including a fully coupled RISMC toolkit, for long term coping studies in order to evaluate FLEX and extended station blackout conditions.

- Compete flooding fragility experiments for mechanical components

- Release beta version of seismic probabilistic risk assessment model.

- Flooding model is validated against an accepted set of data.

- Beta 2.0 release of RELAP-7 with selected separate effects tests for validation data sets, validation of 3D single-phase subchannel, preliminary 3D two-phase (7-equation) subchannel, multi-physics coupling to reactor physics (NEAMS codes Rattlesnake and MAMMOTH).

- Release advanced flooding analysis tool suitable for ocean- and river-based flooding scenarios.

- Compete flooding fragility experiments for electrical components

- Initial demonstration RPV steel embrittlement using a bottoms-up, lower length scale model to capture causal mechanisms of embrittlement.

- Flooding fragility models for mechanical components are validated against an accepted set of data.

- Beta 3.0 release of RELAP-7 with additional validation and full multi-physics coupling, validated 3D two-phase subchannel capability, and implementation of droplet model for BWR SBO scenario, reflood phenomena under LOCA, and PWR feed and bleed process.

- Version 1.0 release of RELAP-7 with validation with selected integral effect tests, demonstration of large break LOCA, and three-field flow model, water, steam, droplets.

FY2019-2020

- Complete seismic experiments for critical phenomena.

- Release beta version 3.0 of Grizzly. This version includes capabilities for modeling selected aging mechanisms in reactor internals.

- Flooding fragility models for electrical components are validated against an accepted set of data.

- Implement risk-informed margins management module in RISMC Toolkit that will perform analyst-augmented evaluation of facility safety to search for vulnerabilities and potential management strategies.

- Grizzly (core internals) is validated against an accepted set of data.

- Complete the demonstration of the margins analysis techniques, including a fully coupled RISMC toolkit, for long term coping studies in order to evaluate FLEX and extended station blackout conditions.

- Ensure development and validation to the degree that by the end of 2020, the margins analysis techniques and tools are the generally accepted approach for safety analysis support to plant decision-making, covering analysis of design-basis events and events within the technical scope of internal and external events probabilistic risk assessment.

- Apply margins analysis techniques to evaluation of spent fuel pool issues.

5.2 Integrated Program List

This section provides additional detail into the Pathway subtasks. The Integrated Program List is separated into following technical task priority order (details are found in Table 5-2):

1. RELAP-7 Development
2. RAVEN Development
3. RISMC Applications
4. Grizzly Development
5. QA and V&V of Tools
6. Code Maintenance

Supporting the technical tasks above is a project management activity. This activity provides the project management aspects to support accomplishing the Pathway objectives and other DOE requirements related to project reporting and oversight.

Table 5-1. RISMC Pathway activities list.

Descriptive Activity Title	Activity Description and Major Deliverables
RISMC Management	Support routine project management activities and new program development tasks, report generation, travel, meetings, and benchmarking
RELAP-7 Development	**2016** • Complete the optimized and validated version of RELAP-7 that provides a tool coupled to RAVEN and to other applications (e.g., aging and fuels modules), used to perform as a balance-of-plant capability for the multidimensional core simulators. • Beta 1.5 release of RELAP-7 with improved closure relationships and steam/water properties, completed LWR 0D components (such as jet pump and accumulator), improved LWR components (1D-2D downcomer, 1D pressurizer, optional steam generator designs such as helical), tightly coupled multi-physics fuels performance (NEAMS code BISON), and single-phase 3D subchannel flow capability. **2017** • Beta 2.0 release of RELAP-7 with selected separate effects tests for validation data sets, validation of 3D single-phase subchannel, preliminary 3D two-phase (7-equation) subchannel, multi-physics coupling to reactor physics (NEAMS codes Rattlesnake and MAMMOTH). **2018** • Beta 3.0 release of RELAP-7 with additional validation and full multi-physics coupling, validated 3D two-phase subchannel capability, and implementation of droplet model for BWR SBO scenario, reflood phenomena under LOCA, and PWR feed and bleed process. • Version 1.0 release of RELAP-7 with validation with selected integral effect tests, demonstration of large break LOCA, and three-field flow model, water, steam, droplets.
Industry Applications	**2016** • Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for enhanced external hazard analyses (seismic and flooding). Release the beta version of initial flooding model. • Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for enhanced external hazard analyses (seismic and flooding). • Develop an initial margin analysis tool to evaluate reactor containment performance to evaluate the reliability of proposed industry BWR hardened venting systems. • Release beta version of seismic probabilistic risk assessment model. **2017** • Complete the ECCS demonstration • Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for reactor containment analysis including hardened reliable vents • Demonstrate margins analysis techniques, including a fully coupled RISMC toolkit, for long-term coping studies to evaluate FLEX for extended station blackout conditions. • Flooding model is validated against an accepted set of data.

Descriptive Activity Title	Activity Description and Major Deliverables
	2018 • Release advanced flooding analysis tool suitable for ocean- and river-based flooding scenarios. **2020** • Complete the demonstration of the margins analysis techniques, including a fully coupled RISMC toolkit, for long term coping studies in order to evaluate FLEX and extended station blackout conditions. • Ensure development and validation to the degree that by the end of 2020, the margins analysis techniques and tools are the generally accepted approach for safety analysis support to plant decision-making, covering analysis of design-basis events and events within the technical scope of internal and external events probabilistic risk assessment. • Apply margins analysis techniques to evaluation of spent fuel pool issues.
RAVEN Development	**2016** • Extend RAVEN to provide an emulator capability for complex systems and an optimization search support for risk-informed margin recovery. • Develop emulator capabilities for complex systems **2017** • Completed software that provides a RAVEN tool that couples to other applications. **2020** • Implement risk-informed margins management module in RISMC Toolkit that will perform analyst-augmented evaluation of facility safety to search for vulnerabilities and potential management strategies.
RISMC Methods Development	**2016** • Demonstrate margins analysis techniques by applying to performance-based Emergency Core Cooling System ECCS) Cladding Acceptance Criteria • Demonstrate the margins analysis techniques, including a fully coupled RISMC toolkit, for enhanced external hazard analyses (seismic and flooding). Release the beta version of initial flooding model. • Complete an investigation into the use of the RISMC methodology to validate traditional PRA models. **2017** • Apply margins analysis techniques to reactor containment analysis including hardened reliable vents and shallow- and deep-water flooding and seismic events. • Complete a full-scope margins analysis of a commercial reactor power uprate scenario. Use margins analysis techniques, including a fully coupled RISMC Toolkit, to analyze an industry-important issue (e.g., assessment of major component degradation in the context of long-term operation or assessment of the safety benefit of advanced fuel forms). Test cases will be chosen in consultation with external stakeholders. **2019** • Apply margins analysis techniques to evaluation of spent fuel pool issues.
RISMC Experiments	**2017** • Compete flooding fragility experiments for mechanical components **2018** • Compete flooding fragility experiments for electrical components

Descriptive Activity Title	Activity Description and Major Deliverables
	2019 • Flooding fragility models for mechanical components are validated against an accepted set of data. • Complete seismic experiments for critical phenomena. • Flooding fragility models for electrical components are validated against an accepted set of data.
Grizzly Development	**2016** • Release the beta version 1.0 of Grizzly. This will include engineering fracture analysis capability for RPVs, with an engineering model for embrittlement, and a modular architecture to enable modeling of aging mechanisms. • Validation and benchmarking of Grizzly will be conducted for reactor metal applications. • Validation and benchmarking of Grizzly for concrete aging will be started. **2017** • Release the beta version 2.0 of Grizzly. This version will include capabilities for modeling selected aging mechanisms in reinforced concrete and for engineering probabilistic RPV fracture analysis. **2018** • Validation and benchmarking of Grizzly for concrete aging will be completed. • Initial demonstration RPV steel embrittlement using a bottoms-up, lower length scale model to capture causal mechanisms of embrittlement. **2019** • Release beta version 3.0 of Grizzly. This version includes capabilities for modeling selected aging mechanisms in reactor internals.
QA and V&V of Tools	**2016** • RELAP-7 will be validated against an accepted set of data. • Validation and benchmarking of Grizzly will be conducted for reactor metal applications. • Validation and benchmarking of Grizzly for concrete aging will be started. **2018** • Validation and benchmarking of Grizzly will be completed. **2020** • Grizzly (core internals) is validated against an accepted set of data.
Toolkit Maintenance and Optimization	**2016-2018** • Support RISMC toolkit including bug fixes and minor updates. • Perform optimization on modules in the RISMC Toolkit • Enhance RELAP-7 by adding engineering-type of closure relations.

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APPENDIX A – Bibliography of Recent RISMC Reports

The current list of reports can be found at:

https://inlportal.inl.gov/portal/server.pt/community/lwrs_program/442/program_documents

Each report link should be associated with the [pdf] text at the end of the row.

Risk-Informed Safety Margin Characterization
Advanced Seismic Soil Structure Modeling, C. Bolisetti and J. Coleman, INL/EXT-15-35687, Idaho National Laboratory, June 2015. [pdf]
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